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SOVIET ATOMIC ENERGY

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(ATOMNAYA ÉNERGIYA)

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SOVIET ATOMIC ENERGY

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FOURTH INTERNATIONAL CONFERENCE OF THE UNITED NATIONS ON THE PEACEFUL USE OF ATOMIC ENERGY

PROSPECTS OF THE DEVELOPMENT OF NUCLEAR POWER IN THE USSR*

A. M. Petros'yants, A. P. Aleksandrov, N. A. Dollezhal', and A. I. Leipunskii

The Fourth International Conference of the United Nations Organization on the Use of Atomic Energy for Peaceful Purposes is being held under conditions in which large industrial atomic power stations are already in operation in a number of countries, and the extensive construction of these is being developed. The seven years which have elapsed since the Third Conference have been years of the intensive and successful economic assertion of nuclear power as a new source of electrical power production. The sixties were mainly characterized by the search for the best types of reactors and their development under conditions of industrial use. We have now embarked upon a period characterized by the practical use of atomic power stations based on a comparatively limited number of types and modified versions of thermal reactors (two or three for the whole country).

The large-scale construction, design, and prospective development of atomic power stations in a large number of countries bear witness to the ever-growing (and in some countries decisive) part to be played by nuclear power, even in the immediate future. Nuclear power forms a vast branch of the whole power industry [1].

The efficiency and economic effectiveness of the use of atomic power stations may already be estimated in terms of a real delivery of power on the industrial scale. The value of atomic power-station development will plainly continue to increase.

No less important is the moral implication of the affair. Advances in nuclear power indicate that this important discovery of the present epoch – atomic energy – may and should be widely employed for peaceful purposes to the benefit of the whole of Mankind.

Factors Determining the Approach to the Development of Nuclear Power in the USSR

The Soviet Union belongs to the group of nations blessed with vast natural power resources. In recent years vast new reserves of coal, oil, and gas have been discovered. The fuel and power basis is being intensively developed. The production of electrical power is increasing annually by 7-8%, and in 1970 it stood at 740 billion kWh. The power of all the electrical power stations in the country has now reached 170 million kW.

Economical thermal power stations with powers of 1200, 2400, and 3600 MW are being constructed. These are equipped with power units of 200, 300, 500, and 800 MW, including those using hypercritical steam parameters. Electrical networks of 400, 500, and 750 kV are being developed. The transmission of direct current at 800 kV has been adopted. Work is proceeding on the creation and adoption of electrical transmission at 1000-1200 kV (ac) and 1500 kV (dc).

The reserves of organic fuel in the Soviet Union would enable us to solve the problem of the further development of energy sources simply on the basis of classical methods. However, there is a certain mismatch in the Soviet Union between the siting of the economic power reserves and the power users.

*This and the articles which follow comprise the contributions of Soviet scientists to the Conference held in Geneva, September 6-16, 1971.

State Committee on the Use of Atomic Energy in the USSR. Translated from Atomnaya Énergiya, Vol. 31, No. 4, pp. 315-323, October, 1971.

The eastern parts of the country have a vast potential of fuel and power and extremely favorable technical and economic conditions for the use of power resources. The power resources of Kazakhstan, Siberia, and Central Asia are particularly great.

The industry of the European part of the USSR and the Urals, which has a material-technological basis, labor reserves, and the necessary raw material, nevertheless experiences a growing deficiency in economic fuel and power resources.

Thus the question as to the scale and tempo of the growth of nuclear power in the power supplies of the Soviet Union constitutes a predominantly economic problem, and is mainly being solved for the European part of the country.

In recent years a number of calculations have been made regarding the changing economic conditions, and many scientific-research and design-engineering projects have been undertaken in connection with perfecting the types of reactors accepted for construction in the USSR for atomic power stations. The results of these investigations have shown that the economic indices obtainable in atomic power stations under the new conditions are in complete accordance with economic requirements.

Atomic Power Stations of the USSR

After the initiation of the world's first atomic power station in Obninsk in the Soviet Union, the development of large atomic power stations, as safe and economically efficient as possible, with a service life of 25-30 years has proceeded with great vigor. The purpose of these developments lay in preparing for the creation of a system of atomic power stations consisting of "first-stage" atomic power stations (using thermal-neutron reactors), providing for the development of electrical power over the European part of the USSR at a lower cost than power stations using ordinary fuel.

Apart from producing electrical power, these power stations also create the fuel required for the future introduction of "second-stage" atomic power stations, based on fast neutrons. These second-stage reactors should possess such a breeding factor as will guarantee the growth tempo needed in the country in order to provide the nuclear fuel for later-developed fast reactors, which in turn should produce power on any desired scale.

In the course of these investigations, the Soviet Union has constructed industrial power stations of various types, using both thermal and fast neutrons, so as to accumulate extensive engineering experience. The design, construction, and practical use of these has yielded a great deal of experience and produced teams of specialists, as well as developing atomic engineering; it has enabled designs to be developed and the construction of large thermal-neutron atomic power stations to be initiated, these forming part of the system of first-stage atomic power stations.

The favorable situation as regards ordinary fuel in the Soviet Union (the development of open-cast mining for very cheap coal in Kazakhstan (Ekibastuz coal basin), the discovery of oil and gas fields in Western Siberia, the use of powerful hydroelectric power resources) has allowed us to develop nuclear power at a moderate tempo, without any unjustified expenses.

Considering the rapid growth in the call for classical electric power in the Soviet Union and the high cost of long hauls of fuel from East to West, it may readily be seen that the economy resulting from the development of nuclear power in the European part of the Union, in the Caucasus, and later in the Urals will continue to increase as the scale of power advances.

Thus in the five years now beginning (1971-1975) it is proposed to make a considerable advance in nuclear power by constructing large power stations incorporating reactors with a specific power of 1 million kW and over, and initiating atomic power stations of 6-8 million kW so as to bring the total atomic-power-station capacity in the next five years to 30 million kW.

After their imminent introduction, the first-stage atomic power stations will mainly work under conditions of basic loading, so as to guarantee the fulfillment of their second function, the preparation of plutonium for use in the second-stage atomic power stations with fast-neutron reactors.

The construction of first-stage thermal reactors will continue until 1980-1985, with a subsequent transition to the construction of atomic power stations with second-stage reactors using fast neutrons. These second-stage atomic power stations, as they come into action, will take on the basic loading, while

TABLE 1. Development of Water-Cooled Water-Moderated Reactors

Characteristic	VVÉR-210	VVÉR-365	VVÉR-440	VVÉR-1000
Power, MW:				
electrical	210	365	440	1 000
thermal	760	1 320	1 375	3 000
Saturated steam pressure in front of the turbine, atm	29	29	44	60
Pressure in reactor vessel, atm	100	105	125	160
Active zone:				
power intensity, kW/liter	46	80	83	111
number of fuel cassettes	343	349	349	151
number of fuel elements in a cassette	90	126	126	331
material of fuel-element cans	Zirconium alloy			
uranium charge, tons	38	40	42	66
enrichment of added uranium, %	2,8	3,0	3,5	4,4
depth of burn-up, MW · days/ton U:				
average	13 000	27 000	28 600	40 000
maximum	19 000	41 000	42 000	44 000
Number of circulating loops per reactor	6	8	6	4
Number and power of the turbogenerators (MW)	3×70	5×73	2×220	2×500

the thermal-neutron reactors will gradually, and as needed, be converted to the half-peak mode of control.

At the moment it is difficult to predict the optimum combination of thermal and fast reactors and the specific extent of the individual stages. These will depend on a number of factors, such as, for example, the cost of uranium and the economics of fast reactors. However, it is vital to ensure that the production of large-scale power and the rate of its growth should be as intensitive as possible to conjectural changes in the price of natural uranium. This will only be possible by having a balanced plutonium fuel cycle, providing for the growth of power by virtue of the fast-neutron reactors at the tempo required for the whole country, i.e., doubling the power in a period of the order of eight years or under.

The foregoing considerations influence the choice of the types of reactors to be used in the Soviet Union.

Atomic Power Stations with Reactors of the Vessel Type

It is well known that the construction of vessel-type reactors, using thermal neutrons, is developing in the direction of one-circuit systems, with the boiling of water in the active zone, and two-circuit systems, with the production of steam in steam generators. In the Soviet Union reactors of this kind have been established in the already-working Melekess atomic power station, in the first and second units of the Novo Voronezh atomic power station, and in the third and fourth units of this station now under construction, in Kol'sk, and elsewhere (Table 1).

Vessel-type reactors are among the most acceptable for practical use; they are distinguished by a high degree of compactness, simplicity of the arrangement, a small amount of construction materials in the active zone, and a relatively low cost. As regards specific power capacity, they are at present among the very best. However, these important advantages are accompanied by certain shortcomings, the importance of which increases with increasing specific power. The first shortcoming is the difficulty of obtaining adequate information enabling the development of defects in the reactor vessel and in the associated large pipes (particularly in regions close to the vessel) to be predicted during the operation of the station. This aspect plays a major part in large atomic power stations in view of the necessity of welding larger and larger parts together during assembly on the site of the power station, rather than under factory conditions.

The neutron losses associated with the necessity of compensating the excess reactivity at the start of a campaign and with absorption in the slags at the end leads to a reduction in the breeding of fuel in the reactor, particularly in the case of deep burn-up. On the whole, the fuel cycle of these reactors is at present hard to control, since the loading frequency of the fuel may only be varied within certain limits, although even here there is a reasonable hope of progress, for example, the use of the plutonium cycle now in prospect, or the use of charges more heavily enriched with uranium.

We have now arrived at the development of the first stage of nuclear power. At the present time we are constructing atomic power stations of 880 MW with two reactors of 440 MW each.

The decision to build these atomic power stations as standard was preceded by prolonged service of the Novo Voronezh atomic power station and a careful revision of its reactors, together with appreciable repair work. In the standard designs now in existence, we have allowed for all the experience so achieved, and at the present time the construction of atomic power stations with these reactors in the center of Russia, in the Kol'sk peninsula, in the Caucasus, and in the Ukraine is technologically and economically viable;

TABLE 2. Development of Power Reactors of the Channel Type with a Graphite Moderator and Ordinary Water as Coolant

Characteristic	BAÉS-1	BAÉS-2	RBMK-1000	RBM-KP-2000
Power, MW:				
electrical	100	200	1 000	2 000
thermal	286	530	3 200	5 620
Steam parameters in front of the turbine:				
pressure, atm	90	90	70	65
temperature, °C	500	500	284	450
Active zone:				
diameter, m	7,2	7,2	11,8	13,5
height, m	6	6	7	7
Number of working channels (including superheating)	998 (268)	998 (266)	1690	1404 (354)
Number of fuel elements in channel	6	6	18	36—37
Mean uranium enrichment, %	1,8	3,0	1,8	2,5
Mean depth of burn-up, MW · days/ton U	4 000	14 600	18 000	24 000
Energy intensity of fuel, MW/ton	4,3	11,3	17,8	19,2
Material of fuel-element cans	Stainless steel	Stainless steel	Zirconium alloy	Superheating channels, zirconium + stainless steel, evaporating channels, zirconium alloy
Number and power of turbines (MW)	1×100	2×100	2×500	2×1000

a few such stations are also being built with the help of the Soviet Union in other countries. The economic indices of these atomic power stations in all the regions indicated are somewhat better than those of power stations using ordinary fuel [2].

It is particularly advantageous in our popular economy to use these atomic power stations in place of the earlier-proposed condensation-type high-parameter power stations based on coal and shale.

The next step in the development of reactors of this type is the construction of atomic power stations with two reactors of 1000 MW each. The first such reactor will be placed in the Novo Voronezh atomic power station, the total power of which will then reach 2.5 million kW.

This reactor will constitute a four-loop installation with a power of 250 MW in each loop.

The vessel and roof of the reactor, like the vessel of the 440 MW reactors, will be made of high-strength heat- and radiation-resistant steel, the properties of which we have studied very fully. The structural changes taking place in these steels under the influence of radiation are far less than in mild steels, since the high-strength steels have a more extended range of ductility and retain a reliable bearing capacity up to integrated irradiation doses an order of magnitude greater than that received in 30 years use of the atomic power station.

As construction material for the active zone we use a high-strength zirconium–niobium alloy (2.5% Nb) having a high corrosion resistance under conditions of irradiation, and for the cans of the fuel elements of zirconium alloy containing 1% niobium.

We do not intend to construct any vessel-type reactors with a power of over 1 million kW in a single vessel. Intermediate powers between 440 and 1000 MW are unnecessary. In this respect there may very well be a gradual transition from six-loop and two-turbogenerator systems to a 500 MW unit with two loops and one turbogenerator. All the equipment of the loops will be made uniform with that of the 1000 MW four-loop vessel-type atomic power station. This will avoid the necessity of developing and running equipment of different types.

It is quite clear that the transition to the unit realization (high-power reactor and turbogenerator) indicated should be accompanied by a general improvement and an increased reliability in the electrical part of power systems incorporating atomic power stations. Up to the present the overwhelming proportion of brief shut-downs in atomic power stations have been due to faults in the electrical rather than the reactor part, and this has to a certain extent decided our choice of one reactor to two turbogenerators. However, this arrangement does slightly increase the cost of the station, which is undesirable as the number of atomic power stations becomes greater; the unit arrangement will therefore come into operation in the future.

Atomic Power Stations with Reactors of the Channel Type Having a Graphite Moderator

The history of the development of power reactors of this type started with the initiation of the First Atomic Power Station in 1954. Later the following were constructed and put into service: in 1958 the Siberian atomic power station (over 600 MW), in 1964 the first and in 1967 the second units of the I. V. Kurchatov Beloyarsk atomic power station (total power 300 MW). The use of the two reactors in the Beloyarsk atomic power station has demonstrated their high radiation safety and reliability. Prolonged operation of the station has demonstrated the possibility of achieving the nuclear superheating of steam on the industrial scale [1].

The channel principle of construction as an alternative to the vessel principle is promising from many points of view; it facilitates the attainment of extremely high specific electrical powers (1000 MW and over), an increase in the parameters of the coolant (and hence the efficiency), flexibility in service and ease in recharging the nuclear fuel, and the absence of cumbersome vessels, which are difficult to construct and transport. These leading features technically and economically justify our intention to introduce powerful reactors of this type into the nuclear power system of the USSR [3].

The next step in the development of the channel principle of construction was the development of the 1000 MW RBMK-1000 reactor, which differs from the reactors of the Beloyarsk atomic power station in the following ways:

1) in the active zone, instead of stainless steel the main construction materials are zirconium alloys, which greatly improve the neutron balance, but require a reduction in the parameters of the coolant;

2) rod-type fuel elements are used in the working channels;

3) the specific power is considerably increased.

In the active zone of the RBMK-1000 reactor the use of the nuclear fuel is considerably improved. The use of the zirconium alloy as construction material, UO_2 as fuel, and graphite as moderator, and also the optimization of the physical and thermotechnological parameters of the reactor, have increased the depth of burn-up of the nuclear fuel to 18,000 MW·days/ton, and the energy intensity of the fuel to 17.8 MW/ton.

The RBMK-1000 reactor is taken as standard in several two-reactor 2000 MW atomic power stations now being constructed. The first of these atomic power stations (Leningrad) is in the stage of equipment assembly [3].

We consider that a leading advantage of this reactor is the fact that the division of the active zone into individual channels of smallish cross section alleviates any hazard associated with a breakdown of airtightness in individual channels or even groups of these. Any defective channel in the active zone may be replaced. In channel-type reactors, service channels of large diameters may be entirely eliminated or brought out to points easily accessible for regulation. Any channel may be discharged with the reactor in operation, without stopping the power station, and the fuel cycle is therefore easily regulated. A disadvantage is the fact that these reactors have more extensively-developed pipelines for the active coolant; they are less compact, require greater construction space, and the capital outlay on them is therefore appreciably higher than in the atomic power stations with vessel-type reactors. However, the possibility of easily rearranging the fuel cycle for purposes of optimization, the fact that the active zone is made up of standard mass-produced parts of low cost and technologically high quality, and also the fact that reactors of practically any power, for example, 2 million kW, may be made from these parts, makes such reactors extremely attractive.

There is still scope for increased compactness in channel-type reactors.

We have thus come to the conclusion that, for specific electric power of about 1 million kW and over, channel reactors are entirely competitive with those of the vessel type. The use of reactors of the type in question has confirmed their great reliability and ease of repair [4].

Since these reactors rest on a broader industrial basis and, as regards their thermal cycle, are more efficient at creating the fuel needed for introducing the second-stage fast-neutron reactors, a large proportion of the Soviet Union's efforts will be devoted toward the creation of channel reactors as well as vessel-type nuclear power stations.

TABLE 3. Principal Characteristics of Fast-Neutron Reactors

Characteristic	BOR-60	BN-350	BN-600
Power, MW:			
thermal	60	1000	1450
electrical	12	350 or 150 MW (el.) and 120,000 m^3/day water	
Depth of burn-up, %	10	5	10
Coolant of first and second circuits	Na	Na	Na
Flow in loop of first circuit, m^3/h	500	3200	9000
Time between rechargings, days	145	50	150
Diameter of active zone, cm	40	160	205
Temperature of sodium at outlet from reactor, °C	2	5	3
Mean energy intensity, kW/liter	800	470	550
Steam parameters in front of turbine:			
temperature, °C	500—540	440	500
pressure, atm	90	50	130

Channel reactors offer the possibility of producing superheated steam, as clearly demonstrated by experience with the two reactors of the Beloyarsk power station.

This possibility should be economically realized after creating zirconium alloys (which have a low neutron absorption) for the channels and casings of the superheating elements. In certain cases, however, for example, in producing high temperature and pressure industrial steam in places with difficult access for the fuel, or in a number of cases in the heating of northern cities, superheating elements with steel casings may also prove desirable, and as need arises such reactors will be constructed, if fast-neutron reactors with high steam parameters are not more economical.

As the next step in the development of reactors of this type, we may consider the design of a reactor with steam superheating of the RBM-KP-2000 type (2000 MW electrical). The characteristics of this reactor are presented in Table 2.

Another possible development is the further improvement of channel reactors, without nuclear superheating of the steam. An increase in the energy intensity of the fuel and the optimization of the physical and thermal characteristics of the active zone should enable us to bring the power of this reactor up to 2000 MW (electrical). This will open the way for a further increase in the economy of atomic power stations with channel reactors.

At the first stage of development, a large proportion of the atomic power stations with reactors of both the vessel and channel types will operate in a mode of basic loading. This is due not only to economic considerations but also the desirability of achieving a maximum supply of fuel for the reactors in the second stage of development of nuclear power.

In regions with the most expensive fuel, in which atomic power stations will naturally predominate, it will gradually become necessary to convert some of the atomic power stations into the regulation mode.

We shall have to make this conversion for those atomic power stations which produce the lowest yield of plutonium, and the operation of these will have to be optimized for 4-5 thousand hours per year. Retaining one recharging per year, this will enable us, for example, to convert the stations to less enriched fuel. It is not impossible that it may yet become desirable to create specialized atomic power stations optimized for 5000 h/year. We may reasonably expect that the need for this will arise no earlier than 1985, when a considerable proportion of the basic loading will be taken on by atomic power stations of the second stage with fast-neutron reactors (Table 3). For these atomic power stations, the use of the uranium–thorium fuel cycle will be extremely attractive.

Atomic Power Stations Using Fast-Neutron Reactors

The dominant quality of these reactors is their more efficient use of the original nuclear fuel and the possibility of attracting U^{238} and also thorium into the fuel cycle; this factor has decided the important role assigned to Soviet fast-neutron reactors in the nuclear power of the future. Historically, thermal reactors became viable far in advance of the more complicated fast-neutron reactors. Many technical and scientific problems have to be solved and experience gained in experimental models before powerful, highly-economic and reliable fast power reactors can be manufactured and taken into service. It is well known that these reactors are particularly efficient for plutonium loading, and they may accordingly very well be taken in conjunction with thermal reactors.

Intensive investigations are taking place in the USSR in relation to fast reactors. A series of experimental fast reactors of low thermal power has been constructed (BR-1, BR-2, BR-3, BR-5, BFS).

After these small experimental reactors based on fast neutrons, the BOR-60 reactor was built in Melekess; this yielded electrical power in 1970. The construction of an atomic power station with a BN-350 reactor is being completed in Shevchenko, together with a desalination plant. The construction of the third Beloyarsk atomic power station with a BN-600 reactor has also begun. The use of these reactors offers the possibility of obtaining the necessary experience and developing the fuel cycle in all its details. The aim is to achieve the tempo needed for our economy by doubling the power of the second-stage atomic power stations in a period of the order of eight years, with very little or no introduction of fuel from outside into the fuel cycle of these atomic power stations. It is quite possible that, at the very beginning, some of the fast reactors will have to be supplied with uranium fuel [5].

The construction of second-stage atomic power stations on the large scale will be carried out as experience accumulates in operating the first atomic power stations, and as the engineering and fuel-cycle capabilities develop. The fast-neutron reactors and their fuel cycle should then reduce the doubling time, first to 8-9 and then to 6-8 years. We have worked out a number of promising ways of achieving this end in concert with the other socialist countries [6]. For this purpose we are developing atomic power stations of 1000-1500 MW, optimized with respect to their technical and economic indices and the breeding of nuclear fuel.

It would appear likely that, on successfully solving the problem of fast reactors, the development of electrical power in the whole of the European part of the Soviet Union will proceed mainly by way of the construction of atomic power stations with fast-neutron reactors from 1985 and onward.

Estimation of the Prospects up to the Year 2000

The successful adoption of large industrial atomic power stations in the USSR and the initiation of a broad program for the construction of these enables us to set out the prospects for the development of nuclear power in the country to the end of this century. Nuclear power is part of a general and unified system of power production; hence in attempting any long-term prediction of its development we must include generalized predictions as to the total power required in the country, the character and structure of the demands made on the system, and the mode of use of the power-generating services.

It is quite clear that any quantitative estimates of the growth in the various sources of electrical power (including nuclear power) over a period of 30 years involve a certain probability factor. Considering the very progress of nuclear power, the rate of change of many operative factors, their magnitude, and their time dependence can hardly be determined accurately over such a long period. The long-term prediction of the development of nuclear power is thus, on the whole, a problem in which existing laws may well change under the influence of as-yet unknown events or factors [7].

The main power reserve of the country will continue to be organic fuel up to the year 2000. Nuclear power will be employed for supplying energy to those parts of the country in which the economic effect of its use will be maximal. According to present predictions, the proportion of total power which will be produced by atomic power stations in the country may well rise sharply over the next few years, so that the construction of new condensation-type thermal power stations in a number of regions of the European part of the USSR will practically cease.

Under the conditions existing in the European part of the USSR, the reserve of economic competitiveness of the new and improved atomic power stations with thermal reactors is fairly considerable. This will justify a certain increase in the specific expenditure on the mining of uranium, provided that the whole economy arising from the effects of other factors leading to an improvement in the economy of the atomic power stations (increased specific power, depth of burn-up, and efficiency, reduction in the expenditure on the manufacture of fuel elements, cheapening of the construction and shortening of the time involved, etc.) will compensate this increase, and the economic competitiveness of nuclear power as a whole will be preserved.

The main strategy governing the development of nuclear power in the USSR is aimed at the all-round and accelerated development and introduction of fast reactors with an expanded breeding of fuel.

As the construction of standard and powerful fast reactors begins, the total power of atomic power stations with thermal reactors in the Soviet Union will reach tens of millions of kilowatts. Hence the atomic

power stations with fast reactors will in fact be incorporated into an already well-developed system of nuclear power. The amount of plutonium which will have accumulated in the thermal reactors by that time, and which will be used for charging the first industrial fast reactors, will ensure the development of the same order of power from the fast reactors. From this time onward, the ratio between the contributions of the thermal and fast reactors to the total power will move in the factor of the latter.

The intensive growth of nuclear-power capabilities may be ensured by introducing extremely large power units in atomic power stations. In this respect, channel reactors would appear the more promising; their specific power in the period under consideration may well increase to 2-3 million kW (electrical) or even more. The contribution of such reactors to the increasing capabilities of nuclear power may very well become greater, since vessel-type reactors will probably have to be restricted to powers of no greater than 1000 MW (electrical), because of the difficulty of transporting them by rail.

As regards the optimum specific powers of fast reactors, this problem will have to be solved with due regard to their vital functional purpose, namely, the supply of plutonium for developing further nuclear power. Considering the extremely large specific powers of the fast breeders, it may prove difficult to place them at the basis of the loading graph of the power systems, as is essential for the purpose in question.

At the present stage, the question as to the standardization and unification of the structural and constructional realizations of the atomic power stations and their integral parts takes on a special significance. A number of specific demands made on the materials and equipment, and the special character of the use, servicing, and repair of atomic power stations, necessitate the further development of specialized engineering, which constitutes a decisive factor in the reliability and economic efficiency of atomic power stations.

In addition to the use of condensation-type atomic power stations, an important contemporary problem is the wide use of nuclear reactors for the production of other forms of output in addition to electrical power: heat, fresh water, chemical products, and cold. Outstanding among these problems are the development of "atomic" central heating and also the production of large quantities of fresh water by the distillation of salt water, using reactors as a source of heat.

The successful use of the first nuclear power stations and their freedom from radiation hazard [8, 9] create confidence in the possible siting of atomic thermal power stations close to large cities. Apart from economy in the length of the heat pipelines, this simultaneously leads to the solution of another problem, that of combating the contamination of the air and surrounding territory with combustion products. The contamination of the air in large cities, arising largely from the ordinary thermal power stations supplying them with heat, not only saps the health of the people but also changes the face of Nature. Atomic condensation-type or central-heating power stations eliminate such contamination of the air, while still ensuring radiation safety. From this point of view, the best technological principle lies in the construction of the working channels of the Beloyarsk atomic power station. The construction of the fuel elements of this atomic power station, as long service experience confirms, eliminates the fall-out of radioactive fission products into the circuit and ensures an excellent radiation environment both inside and outside the atomic power station.

Another extremely important popular-economy problem, the production of fresh water from salt, using the heat from nuclear reactors on the industrial scale, is already beingsolved inthe Soviet Union by the erection of a large atomic power station in the city of Shevchenko. This city lies on the eastern shore of the Caspian sea, where there are no natural sources of fresh water. With the starting of this atomic power station, the water supply for the city and the surrounding region has been vastly improved. Here we have what is essentially the development of a large agroindustrial complex based on the use of atomic energy.

A study of the prospects of the development of nuclear power shows that the extensive construction of atomic power stations is desirable, not only because of their higher economy relative to that of stations working on ordinary fuel, but also (and no less important) because the extremely high calorie content of nuclear fuel leads to a substantial economy in the labor resources occupied in the mining of organic fuel and its transportation.

The development of the popular economy of the USSR in the long-range view (up to the year 2000) will demand such an expenditure of fuel and power resources as will necessitate a radical change in the structure of the incoming part of these resources.

The basis of this change may well be nuclear power, which by the year 2000 will have been converted into a fundamental branch of the fuel and electrical-power economy of the country.

It is precisely these questions which will determine the scale, the prospects, and the methods of development of nuclear power in the Soviet Union in the period culminating at the end of the century.

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DEVELOPMENT OF ATOMIC POWER PLANTS WITH WATER-MODERATED, WATER-COOLED REACTORS IN THE SOVIET UNION

V. P. Denisov, Yu. V. Markov,
V. A. Sidorenko, S. A. Skvortsov,
V. V. Stekol'nikov, and L. M. Voronin

Atomic power plants with water-cooled, water-moderated reactors are widely used in the nuclear power system of the Soviet Union. Several installation with such reactors have been developed: the VVER-210, VVÉR-365, VVÉR-440, and VVÉR-1000 (the number gives the gross electrical power of the reactor in megawatts).

Chronologically and technologically, we can divide these reactors into three generations: a first generation consisting of the experimental-production VVÉR-210 and VVÉR-365 reactors, although the latter occupies a somewhat special position, characterizing the transition from the first generation to the second; a second generation, consisting of several versions of the moderate-power VVÉR-440 serial-production reactor; and a third generation consisting of the high-power VVÉR-1000 serial-production reactor.

Paths toward the Improvement of Water-Cooled, Water-Moderated Reactors

The VVÉR-210 and VVÉR-365 reactors have been in use as the first and second units of the Novo-Voronezh Atomic Power Plant (NVAPP) since 1964 and 1969, respectively. The VVÉR-70 reactor, constructed by the USSR in the German Democratic Republic for the Rainsberg Atomic Power Plant, and in use since 1966, may be added to the list of first-generation reactors. The second-generation VVÉR-440 (Fig. 1) are used in the third and fourth units at the NVAPP; the Kol'skii and Armenian Atomic Power Plants; the plants built in the GDR, Bulgaria, Finland; and in those planned in other countries. The reactor of the third unit of the NVAPP is currently near the end of construction. The third-generation reactor VVÉR-1000 was designed for use in planned atomic power plants. The main unit, with a VVÉR-1000 reactor (Fig. 2), is planned for construction at the NVAPP (fifth unit).

The basic considerations involved in the design and construction of each reactor were to make use of factory-built vessels and railroad transportation. Factory manufacture largely satisfies the requirements for quality and reliability over long periods of use. When railroad transportation is feasible, atomic power plants can be built in many regions of the USSR and in other countries.

The vessel size used in the very first unit of the NVAPP (VVÉR-210) was essentially the maximum permissible for railroad transportation, so the transition to higher powers has primarily involved more efficient use of the vessel volume. Improvements in the design and manufacture accompanied an increase in the working pressure and a corresponding increase in the efficiency of the thermodynamic cycle; along with an increase in the size of the basic equipment of the installation, this has resulted in increased productivity of the atomic power plants.

Table 1 shows the basic technical characteristics of the reactors.

Experience gained in the operation of the first unit of the NVAPP [1, 2] showed that the possibilities of the active zone of this size (H = 2.5 m, D = 2.9 m) were far from being exhausted. It was shown [3] that

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improvement of the active zone by means of measures taken to balance the neutron field permit 75% more power in the same volume, i.e., an increase of the active-zone power to 1300-1400 MW (thermal).

In the design of the second unit of the NVAPP (VVÉR-365), it was decided to proceed on the basis of proven engineering solutions and basic manufacturing equipment in order to accelerate the construction of the reactor and thus more rapidly gain experience in the operation of an active zone with a high electrical power. The basic principles and methods of implementing the engineering decisions adopted regarding the construction of this reactor are described in [3, 4].

While the second unit was being designed, work was begun on the development of the VVÉR-440 reactor, intended to serve as a "serial-production" version for the construction of several atomic power plants. This installation uses an active zone of essentially the same power as in the second unit, but the electrical power is higher because of an increased overall plant efficiency (to 32%). The increase in the saturated-vapor pressure before the turbine which is required here is achieved by increasing the working pressure in the reactor vessel to 125 kg/cm^2 (Table 1). In addition, the power equipment was increased in size, by means of two 220-MW turbogenerators. The VVÉR-365 and VVÉR-440 reactors, based on the VVÉR-210, despite some differences in technical indices, incorporate the same fundamental solutions regarding the active zone, including the reactivity-compensation system, and regarding the basic reactor equipment.

The VVÉR-1000 reactor is based on slightly different fundamental engineering solutions. This high power is achieved by means of the following measures: a) there is more efficient use of the vessel volume because of a change from the reactivity-compensation method used for the preceding reactors, based on "two-story" control mechanisms (the fuel, linked with the neutron trap-absorber), to "single-story" clusters of absorber rods combined with a chemical control method. This permits an increase in the size of the active zone within a vessel which can be transported by railroad; b) a further increase in the specific electrical power of the active zone is achieved through an equalization of the neutron field and an increase in the degree of fuel burn-up; c) the characteristics of the primary coolant are improved, and there is a simultaneous increase in the unit power of the basic equipment in the first circuit through the use of highly efficient pumps whose rotating parts have a high inertia; d) powerful turbogenerators are used with more efficient vapor parameters.

The fuel used in all these reactors is baked uranium dioxide having a density of about 10.4 g/cm^3, clad in tubular jackets of a zirconium alloy with 1% niobium equal in length to the height of the active zone. In all the reactors the active zone is made from hexagonal cassettes. The reactor of the second unit of the NVAPP and all subsequent reactors use fuel elements having an outside diameter of 9.1 mm with a core 7.55 mm in diameter (in the first unit, the fuel elements have an outside diameter of 10.2 mm); these elements were developed in connection with the production of the VVER-365 reactor in order to increase the size of the heat-exchange surface in the active zone.

The reactors operate with three partial fuel rechargings per run. In all the reactors except the VVÉR-210, the recharging procedure always involves positioning of the fresher fuel at the periphery of the active zone and its subsequent movement to the central region of the zone (where it stays for two working periods), from which it is ultimately discharged. This procedure of moving the fuel results in the necessary equalization of heat evolution in the active zone; combined with measures to increase the heat-exchange surface area, it produces the necessary thermal power of the active zone while maintaining the thermal parameters within safe limits. In addition, the difference between the average and maximum consumption of the discharged fuel is reduced. The working periods between rechargings are planned to be 6500-7000 eff.h; this will result in efficient use of the plant power and will permit recharging once annually, during the spring-summer period convenient for power systems. (Table 1) shows the particular enrichment, loading, and consumption values adopted.

The procedure by which the fuel is moved during a recharging from the periphery to the center results in increased breeding at the edge of the active zone and increased danger that criticality will result at the periphery of the reactor in the cold state. Combined with the increased initial reactivity reserve, this circumstance forces an increase in the compensation capability of the control and shielding system, at the transition from the active zone of the VVÉR-210 to subsequent zones.

The compensating capability of the absorber system with a water trap for fast neutrons is governed primarily by the absorber size and spacing in the active zone. In order to suppress excess reactivity at the periphery of the active zone by means of mechanical absorbers, a grid of absorbers is arranged at

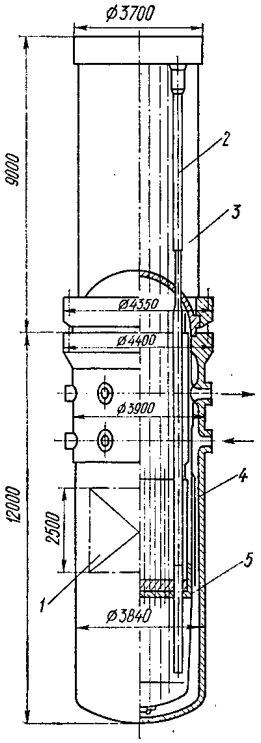


Fig. 1

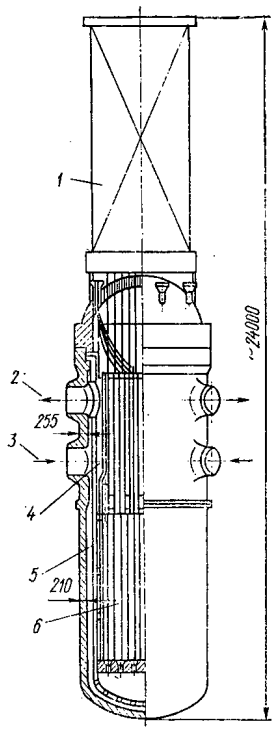


Fig. 2

Fig. 1. The VVÉR-400 reactor. 1) Active zone; 2) automatic control actuator; 3) upper unit; 4) frame; 5) shaft.

Fig. 2. The VVÉR-1000 reactor. 1) Upper unit with actuators for safety and control rods; 2) coolant outlet; 3) coolant inlet; 4) shielding-tube unit; 5) shaft; 6) active zone.

the edge of the reactor; with the fuel cassettes and absorbers having the same size in the VVÉR-365 and VVÉR-440 as in the VVÉR-210, the result is a system of 73 control mechanisms placed uniformly through the active zone at nodes of a triangular lattice with a spacing of 294 mm. The system of such absorbers completely immersed in the active zone is efficient enough to maintain a cold unpoisoned reactor in the subcritical state, without the use of additional absorption means.

The uniformity of these regulators and the absence of special fuelless rods as emergency apparatus permit complete use of the entire active zone, an increase in the heat-exchange surface, an increase in the charging of the reactor with fuel, and an improvement in the neutron balance in the active zone.

This led to the design of control mechanisms of standardized capabilities, which achieve emergency shut-down of the reactor through independent downward motion of an absorber along with fuel at a high velocity when a voltage is supplied from an electrical lead. However, the increase in the multiplying properties of the active zone and the high efficiency of each control mechanism increase the danger that there will be an increase in excessive reactivity when a single absorber is introduced. For this reason, the control-mechanism actuators have a provision for the introduction of emergency shielding from any vertically intermediate absorber position in the active zone.

The distortion which these absorbers cause in the heat-evolution field is essentially removed when a solution containing boron is used in the coolant for the first circuit; in this manner, nearly all the absorbers can be removed from the active zone during power operation.

In the second unit of the NVAPP, boron is used in the primary coolant to compensate for the burn-up reactivity reserve, so that experience can be gained in operation with chemical control of the reactivity. The positive results obtained have resulted in the widespread use of this reactivity-control method in recent versions of the VVÉR-440, and the number of absorber traps has been reduced to 37 through a uniform distribution of these traps at a spacing of 441 mm in the active zone. A cold unpoisoned reactor in this case

TABLE 1. Basic Characteristics of the Water-Cooled, Water-Moderated Reactors

Characteristic	VVÉR-210	VVÉR-365	VVÉR-440	VVÉR-1000
Power, MW:				
electrical	3×70	5×73	2×220	2×500
thermal	760	1320	1375	3000
Overall efficiency, %	27,6	27,6	32	33
Vapor pressure before turbine, abs. atm	29	29	44	60
Pressure in circuit I, abs. atm	100	105	125	160
Number of circuits	6	8	6	4
Water flow rate through reactor, m^3/h	36 500	49 500	39 000	76 000
Water temperature at inlet to reactor, °C	250	250	269	289
Average heating in reactor, °C	19	25	31	35
Active zone				
Equivalent diameter, m	2,88	2,88	2,88	3,12
Height, m	2,50	2,50	2,50	3,50
Number of fuel cassettes	343	349	349	151
Diameter of rod fuel element, mm	10,2	9,1	9,1	9,1
Number of rods per cassette	90	126	126	331
Spacing of fuel-element grid, mm	14,3	12,2	12,2	12,6
Uranium charge, m	38	40	42	66
Average steady-state degree of fuel burn-up, MW·day/(kg U)	13	27	28,6	26-40
Average operating time between fuel recharging, eff. h	5200	6500	7000	7000
Average enrichment of make-up fuel in steady state, at.%	2,0	3,0	3,5	3,3-4,4
Average specific electrical power of active zone, kW/liter	46	80	83	111
Average line load, W/cm	97	125	131	176
Initial reactivity reserve at 20° C, %	11,5	13,8	18	22,5
Number of control mechanisms	37	73	37	109
Overall efficiency of the control mechanisms at 20° C, %	14	20,6	10	6,2

is brought into the subcritical state by means of a boron solution in the water of the first circuit. Two considerations were involved in the construction of the mechanical control mechanisms in the VVÉR-1000.

First, the need to fit the 3000-MW (thermal) active zone into a vessel transportable by railroad forces a more efficient use of the vessel height. The result was that movement of fuel assemblies could be avoided, and a transition could be made from a "three-story" zone (in which the lower "story" is used to position the fuel assemblies moved out of the active zone) to a "two-story" zone. This permitted an increase in both the height and diameter of the active zone.

On the other hand, the increase in the coolant velocity which accompanied the increase in the reactor power under these conditions complicates the use of control mechanisms with movable fuel because of the ejection tendency of the water flow.

As a result, control mechanisms without displacement devices in the form of sheafs of absorber rods are used; these mechanisms are placed in essentially each heat-evolving assembly and are introduced into special guide tubes.

A boron-containing solution is used in the coolant to compensate for the slowly changing effects of the reactivity; it is also used to bring a shut-down reactor into the subcritical state.

Basic Structural Features of the Reactor Equipment and Evolution of these Features

In addition to factory manufacture and railroad transportation, the following solutions were adopted in the construction of the reactors and are common to the reactors of all three generations.

1. High-strength, low-impurity steel is used for the reactor vessel; this results in minimum size and weight and thus makes the railroad transportation possible.

2. The reactor vessel is made from seamless-forged shells, without longitudinal welding seams, so that operating reliability of the reactor is increased.

3. The lower part of the vessel, where the active zone is placed, is an integral cylindrical shell with an elliptical bottom, without any kind of aperture; the inlet and outlet fittings for the primary coolant pipes and for other purposes lie no less than 1 m above the upper part of the active zone. This results in increased strength, because there are no additional stress concentrations, and the active zone is reliable contained in emergency situations due to supercondensation in the first circuit.

TABLE 2. Basic Characteristics of the Steam Generators

Characteristic	NVAPP unit I	NVAPP unit II	Plant with VVÉR-440	Plant with VVÉR-1000
Thermal power, kcal/h	$109 \cdot 10^6$	$154 \cdot 10^6$	$195 \cdot 10^6$	645·1
Steam production rate, metric tons/h	230	325	452	1469
Vapor pressure at outlet, abs. atm	32	33	47	64
Temperature of supply water, °C	189	195	226	220
Temperature at coolant inlet, °C	273	280	301	322
Temperature at coolant outlet, °C	252	252	268	289
Coolant velocity, m/sec	2,94	3,36	2,7	4,89
Temperature drop, °C	24,7	25,5	21,2	24,7
Average heat flux, kcal/m²·h	$91 \cdot 10^3$	$95,5 \cdot 10^3$	$80 \cdot 10^3$	158,1
Number of pipes	2074	3664	5146	15648
Pipe diameter and wall thickness, mm	21×1,5	16×1,4	16×1,4	12×1,5
Vapor moisture (average calculated), %	0,001	0,0038	0,005	0,2
Weight of dry steam generator, metric tons	104,2	112	145	265
Metal expenditure, metric tons of metal per metric ton of steam	0,45	0,344	0,32	0,18

4. The devices within the vessel are removable for easier repair and replacement and for easier monitoring of the interior surface of the vessel.

5. The control and safety apparatus and that for monitoring the reactor operation (the water temperature at the outlet from the active zone and the energy evolution) is placed in the upper part of the reactor for convenient servicing, repair, replacement, etc.

6. The vessel support is placed in the central part of the reactor, near the plane of the axes of the primary circulation pipes, for minimum thermal deformation (or displacement) when the reactor equipment is converted from the cold to the hot state. The vessel support is an annular shoulder which is in integral part of the vessel wall.

7. The bottom of the vessel is elliptical for easier manufacture from a single seamless-forged piece; the result is more reliable operation because there are no longitudinal welding seams.

8. Throttle cylinders with perforated walls are mounted ahead of the outlet fittings where the coolant leaves the reactor, for optimum distribution of the coolant flow and thus suppression of structural vibrations under the removable cover.

9. The apparatus within the vessel is placed in a cylindrical shaft fixed at the top and restrained from radial or circular motion at the bottom. This shaft separates the inward and outward coolant flows in the reactor.

10. The fuel cassettes are placed in a removable basket with a thick-walled supporting grid which fixes and correctly positions the lower parts of the fuel cassettes. At the top, the cassettes are clamped by devices with supporting grids in which there are remote-control devices for correct positioning of the tops of the fuel cassettes.

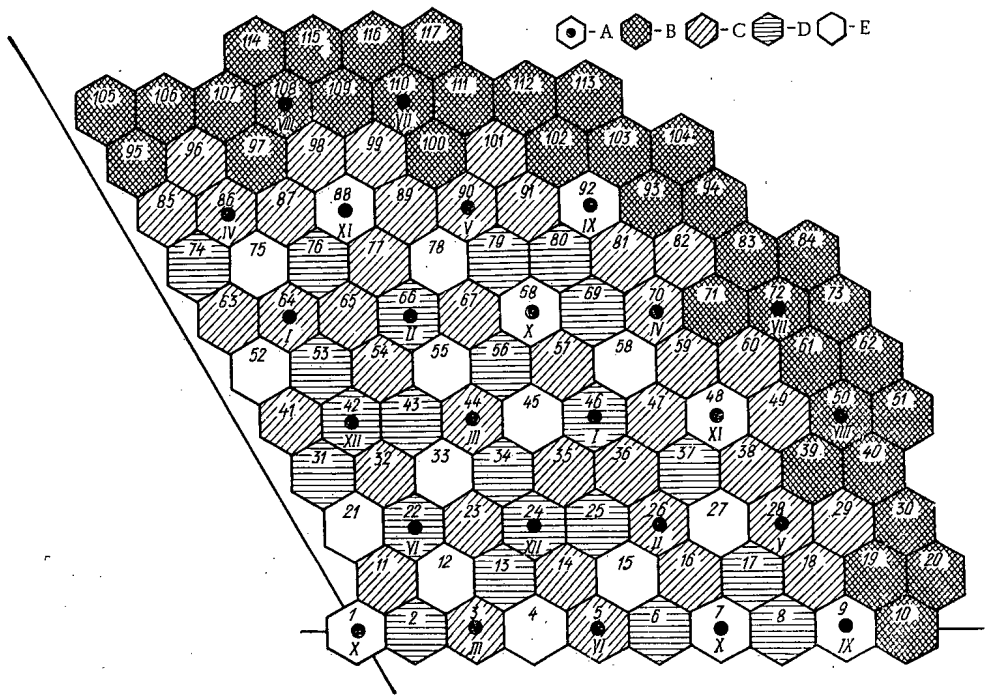
11. The coolant temperature at the outlet from the cassettes of the active zone is measured by means of replaceable temperature pickups placed in "dry" channels on the reactor cover.

In addition, the reactors in each generation have distinctive features resulting from the operating conditions, the parameters of the apparatus, and from the experience gained in design and operation. We will give two examples of these features. The first-generation reactors VVÉR-210 and VVÉR-365 are designed for a first-circuit working pressure of 100 and 105 kg/cm^2, respectively, so a flat vessel cover 500 mm thick with self-sealing wedge-shaped closing devices can be used. A wedge-shaped nickel or copper spacer 3350 mm in diameter is used. The wedge-shaped closing devices require the use of a band of the vessel flange.

Experience has shown that these closing devices provide a hermetic seal and are convenient to assemble and disassemble.

Spherical covers are used in the VVÉR-440 and VVÉR-1000 reactors since at first-circuit pressures of 125 and 160 kg/cm^2, flat covers must be made quite thick and are thus difficult to use.

The use of spherical covers led to the design of seals by means of spacers, which do not cause any additional stress and which do not require a band at the neck of the vessel.



	U^{235} enrichment, %		
	3	2	1,5
No. of working cassettes	96	93	51
No. of control cassettes	12	27	15
Total number of cassettes	108	120	66

Fig. 3. Map of the first fuel charging of the active zone of the VVÉR-365 reactor of the NVAPP (symmetry sector). A) Control cassette (the Roman numerals indicate the group of control cassettes); B, C, D, E) U^{235} enrichment of the cassettes of 3, 2, 1.5, and 1%, respectively.

Mechanisms with a kinematic pair consisting of a screw and a ball nut are used to displace the control rods in the VVÉR-210 and VVÉR-365 reactors. The rods and mechanisms are cooled by first-circuit water, itself cooled and purified in special water-purification devices.

A kinematic pair consisting of a rack and pinion is used in the VVÉR-440 reactor, and the rods and mechanisms are cooled by intermediate-circuit water in a special system not connected to the first circuit. This arrangement makes the operation of the water-purification system used to maintain certain water parameters in the first circuit independent of the operation of the control and safety mechanisms.

The VVÉR-1000 reactor, which has lightweight control devices, does not require powerful rods and mechanisms, as do the first- and second-generation reactors. Lightweight rods can thus be used. The design of the VVÉR-1000 incorporates electromagnetic step-type and plunger drives with air cooling.

The Steam Generators

All the atomic power plants with water-cooled, water-moderated reactors in the USSR use single-vessel horizontal steam generators with submerged heat-exchange surfaces and built-in separation devices.

The following conditions were governing in the choice of steam generator: 1) transportability, i.e., the possibility of transporting the completely assembled steam generator by railroad to the atomic power plant; 2) the possibility of complete assembly and testing at the factory; 3) minimum amount of assembly; 4) reliable heat-engineering and separation characteristics with a minimum hydraulic resistance; 5) reliability of the structural units and of the steam generator as a whole; 6) reliable operation, convenient servicing, and possibility of repair; 7) most satisfactory installation in the plant building; 8) minimum construction cost.

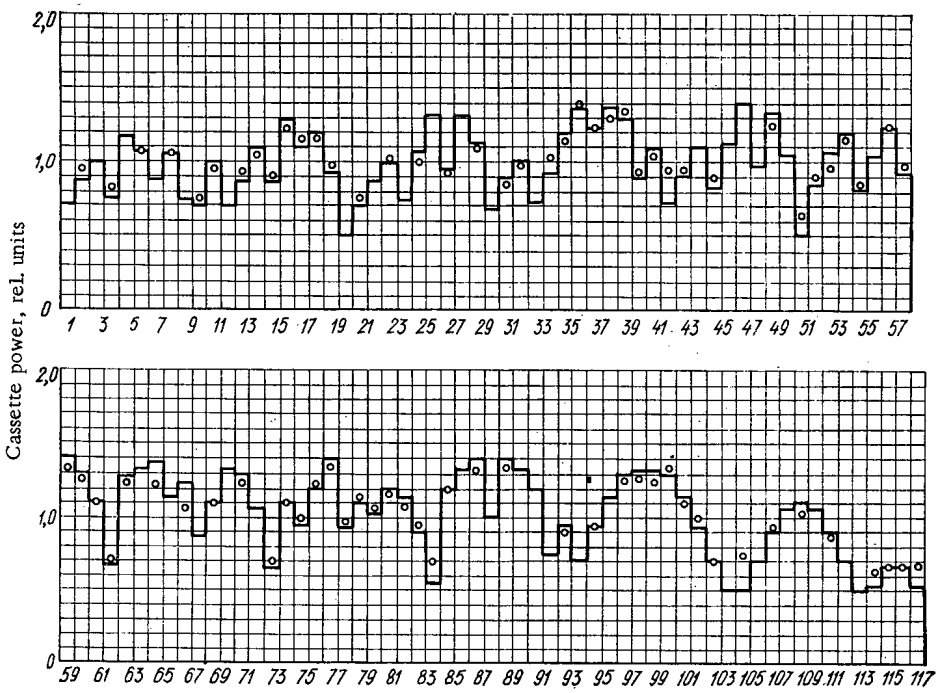


Fig. 4. Comparison of calculated and experimental values (o) of the cassette power of the VVÉR-365 reactor of the NVAPP under operating conditions. Control groups of cassettes are Nos. 24 and 42, at a height of 125 cm. There is 0.36 g boron/kg water in the coolant. The reactor power is 800 MW (thermal). The cells of the symmetry sector are numbered in accordance with Fig. 3. The rms measurement error is ±10%.

These conditions are satisfied best by a horizontal steam generator with pipe boards in the form of circular collectors. This type of steam generator has been constantly improved in the USSR as a result of experience gained with the first and second units of the NVAPP. Table 2 shows the characteristics of the steam generators used for the VVÉR-210, VVÉR-365, VVÉR-440, and VVÉR-1000.

Six steam generators each having a steam production rate of 230 metric tons/h have been operating in the first unit of the NVAPP since 1964. The second unit of the NVAPP has eight steam generators each having a steam production rate of 325 metric tons/h, of the same size and construction as the steam generators used in the first unit.

A plant with a VVÉR-440 reactor has six steam generators each having a production rate of 450 metric tons/h and a design analogous to that of the generators used in the first and second units of the NVAPP, but differing in the method used to service the interior cavities of the pipe collectors, where the sealed ends of the heat-exchange pipes are placed. Hatches are provided in the collectors below the steam generator for inspection or repair in the first and second units of the NVAPP, while the structural part of the plant has special underground enclosures for the equipment for remote control of it. In the steam generators of the VVÉR-440, the hatches on the collectors are on top, and servicing can be carried out directly from the central room of the reactor building; this considerably simplifies both the construction and operation of the steam generator with regard to sealing, unsealing, inspection, and repair.

Various types of construction and technological schemes were considered in the design of the steam generators for the VVÉR-1000 reactor, and the efficiencies of these modifications were compared. A horizontal steam generator was adopted for use in the VVÉR-1000 on the basis of the experience accumulated in the design and operation of saturated-vapor steam generators in the USSR. Structurally it is similar to the preceding generators, differing from them by a significant increase in the inside diameter of the vessel (4000 mm instead of 3200 mm for the VVÉR-440), a decrease in the diameter of the heat-exchange pipes (to 12 mm), and more efficient separation devices. In the steam generators of all the reactors the heat-exchange pipes are sealed into the pipe collectors by roll-in over the entire collector thickness, through the use of the energy of explosives; there is a seam at the collector end of the pipes.

Primary Circulation Pumps

The cooling circuits of the VVÉR-210, VVÉR-365, and VVÉR-440 reactors use hermetic gasketless pumps having a pumping rate of 5600-6500 m^3/h. The use of gasketless pumps with a low inertial mass places stiffer requirements on the system ensuring reliable power supply to the pump motors under emergency conditions due to current cutoff. For this reason, the VVÉR-1000 design incorporates primary sealed-shaft circulation pumps having a large inertial mass. This simplifies the system for ensuring reliable power supply to the pump motors, since quite a long time is available for switching the pump from one power supply to another during current interruptions or for safe shut-down or power reduction of the reactor. The primary circulation pumps of the VVÉR-1000 have a pumping rate of 19,000 m^3/h. The use of more powerful pumps permits the number of circulation circuits servicing the reactor to be reduced to four.

Primary Circulation Pipes

The production of seamless-forged pipes with an inside diameter of 500 mm and a wall thickness of 25-35 mm of Kh18N10T stainless austenite steel was optimized during the design and construction of the first group of atomic power plants having water-cooled, water-moderated reactors. Accordingly, essentially the same dimensions and materials were incorporated in the design of the VVÉR-210, VVÉR-365, and VVÉR-440 for the primary circulation pipes. Pipes having an inside diameter of 850 mm, stamp-welded from carbon steel 55-75 mm thick and clad inside with a stainless steel layer ~5 mm thick, will be used in the VVÉR-1000, in which the basic circulation-circuit apparatus has been considerably enlarged.

Shut-off valves are placed in the primary circulation circuits; there is one in both the "cold" and "hot" branches of each circuit.

A distinctive feature of the valves is a closing device with two disjoining "plates," which provide a seal in two independent planes. After the valve is closed, water is injected into the region between the plates at a pressure greater than that in the reactor; the result is a reliable seal of the plates on both the reactor and steam-generator sides. The valve providing a seal in this manner is essentially equivalent to two ordinary valves.

An electric drive is used for all the valves. Experience has shown that shut-off valves result in more convenient plant operation.

Volume Compensators

All the plants with water-cooled, water-moderated reactors, except for the VVÉR-210, employ a vapor system for compensating for the coolant volume. The VVÉR-210 in the first unit of the NVAPP has a gas compensation system in which the volume compensators are four separate vessels having a total volume of 68 m^3. The vapor compensators in the second unit of the NVAPP, on the other hand, consist of four separate vessels having an inside diameter of 1500 mm and a total volume of 43 m^3, because of engineering considerations. The vapor volume compensator in the VVÉR-440 is a single cylindrical vessel having an inside diameter of 2400 mm and a volume of 38 m^3.

A vapor volume compensator also in the form of a single vertical cylindrical vessel, with an inside diameter of 3000 mm and a volume of 77 m^3, is planned for the VVÉR-1000.

The volume compensators are made primarily from carbon steel - alloyed steel for the VVÉR-1000 and ordinary boiler plate for the others. Both the gas and vapor compensation systems have operated successfully in the first and second units of the NVAPP.

Recharging System

A recharging process involving the transport of burnt-up fuel by a recharging machine under a protective layer of water was successfully incorporated in the VVÉR-210.

Mechanical devices were initially used for emergency shut-down of the reactor during recharging in the event of excessive charging. During the third recharging in 1967 these devices were replaced; boric acid was added to the water in a concentration which would prevent criticality in the event of possible recharging errors. After the boric acid system proved successful in the VVÉR-210 reactor, it was incorporated in all designs.

Reconstruction of the VVÉR-210 Reactor

The first unit of the NVAPP, with a VVÉR-210 reactor, was in constant operation for more than 5 yr, producing about $7 \cdot 10^9$ kWh of electrical power in December, 1969 [1, 2]. In order to examine the reactor after this 5-yr operation and in order to carry out planned modifications in the active zone, it was decided to carry out a complete examination of the VVÉR-210 reactor, with extraction of the fuel and apparatus within the vessel. The shut-down was planned to occur after the second unit of the plant was put into operation, but it actually came somewhat earlier, in early December, 1969. The shut-down came early because of abnormal indications of the instrument measuring the pressure drop in the reactor; this was evidence that a pipe for pressure take-off had failed. This pipe was structurally integral with the heat shield of the reactor vessel and could have been destroyed only as a result of a displacement of the shield.

After complete removal of the fuel and the reactor shaft, it was discovered that failure of the heat-shield support had allowed the shield to drop, and its bottom was resting in the elliptical bottom of the vessel. Some parts of the apparatus holding the screen on the vessel were torn off. The shield itself was essentially undamaged, except for a few cracks at the top where brackets for mounting on the vessel were welded. It was difficult to reliably remount the heat shield on the vessel because of the high radioactivity. A refined calculation of the operating capability of the vessel over its planned working life with increased wall irradiation showed that the VVÉR-210 reactor could be used without the shield. The shield was accordingly cut up and removed.

Analysis of the destruction of the heat-shield mount showed that the cause was an inadequate account of the hydrodynamic and vibrational load in the design of the mount.

A new shaft and shaft bottom were put in place in accordance with the modification plan for the apparatus inside the vessel. The purpose of these plans was to make the active zones of the VVÉR-210 and VVÉR-365 reactors the same.

The shaft, shield, and mounting elements were removed by remote control under water with visual monitoring by means of underwater lamps and television cameras.

The weld seams were monitored and slight damage was corrected by means of a shielded container lowered into the internal cavity of the reactor. Equipment necessary for this work was designed and built.

In the course of the reconstruction, an additional study was made of the hydrodynamics of the water flow in the reactor and the vibration resistance of the apparatus inside the reactor. The study resulted in an improvement of this apparatus and the mount of the reactors in subsequent versions, which are accordingly more reliable.

For this work to be carried out, the first unit of the NVAPP was shut down for slightly more than 1.5 yr.

Useful experience was gained in the reconstruction of the VVÉR-210 and incorporated in corrections to the design of water-cooled, water-moderated reactors, whose long-term operation is thus more reliable. It was also demonstrated that repair can be made in a quite complicated radiation situation.

Some Results of a Study of the Active Zone of the VVÉR-365 (Second Unit of the NVAPP)

The first active zone having a high specific electrical power, the active-zone prototype for VVÉR-440 reactors, was put into operation in the second unit of the NVAPP. For this reason, a study of this active zone and its operating characteristics seem clearly of interest not only because of the operating characteristics themselves, but also for comparison of the predicted and actual characteristics.

Many critical experiments were carried out during the start-up tests of the second unit of the NVAPP. Studies were made of the critical positions of the control mechanisms, the efficiency of the boric acid, the energy-evolution distribution in the active zone, the effects of the reactor reactivity, the reactivity coefficients, etc.

The first fuel charging of the reactor had the following composition:

No. of cassettes	U^{235} enrichment, %
55	1,0
66	1,5
120	2,0
108	3,0

Figure 3 shows the cassette arrangement in the active zone and the distribution of control devices among groups. In the 3.0%-enrichment cassettes, there were 120 fuel elements and six rods with expended absorbers acting as an additional means for compensating the initial reactivity reserve. Provision was made for raising any group of control devices without turning the cold reactor critical in the absence of boron in the coolant; accordingly, all the experiments and operations in which chemical control of the reactivity was used were carried out under conditions simplifying the nuclear-safety requirements.

Experimental data [5] on the change in the critical positions of the control mechanisms during the first reactor run, carried out without boron in the coolant, agree within ±0.002 with the calculated data for k_{eff} over the entire temperature range. The calculations were carried out by a program analogous to that described in [6].

The efficiency of eight control groups (VI-XII) measured at 100°C is $\Delta k/k = (11.60 \pm 0.30) \cdot 10^{-2}$, in good agreement with the calculated value of $11.38 \cdot 10^{-2}$.

The temperature coefficient of the reactivity is essentially independent of the temperature, in agreement with the conclusion reached from the calculations. The reason for this is that under the experimental conditions, a decrease in the temperature in the reactor was accompanied by an increase in the number of control absorbers in the active zone, i.e., by an increase in the neutron leakage. In the working range of water temperatures, the temperature coefficient of the reactivity for boron-free water was $4.3 \cdot 10^{-4}$ deg^{-1}. A power coefficient of the reactivity in good agreement with the calculated value of $1.5 \cdot 10^{-4}$ (%)$^{-1}$ was found from the temperature coefficient in experiments checking the reactor self-control.

Considerable attention was paid to the effect of the boron in the first-circuit water on the characteristics of the active zone. The efficiency of the natural boron in water at 100°C [$(18.5 \pm 0.5) \cdot 10^{-2}$ g/kg] found in critical experiments was exactly equal to the calculated value. The effect of boron on the temperature coefficient of the reactivity was found in experiments on reactor self-control. With a boron concentration of 0.5 g/kg in the water, e.g., the temperature coefficient of the reactivity decreased to $1.75 \cdot 10^{-4}$ deg^{-1}, but this caused essentially no decrease in the stability of the reactor with respect to perturbations in the steam load.

The distribution of power among the active-zone cassettes was studied in measurements of the water heating in the cassettes. As Fig. 4 shows, the measured results for most of the cassettes agree with the calculated values within the experimental error.

The experimental information offers further support for the suitability of the basic physical characteristics planned for the active zones of the VVÉR-440 reactors.

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THE LENINGRAD NUCLEAR POWER STATION AND THE OUTLOOK FOR CHANNEL TYPE BWR'S

A. M. Petros'yants, A. P. Aleksandrov,
N. A Dollezhal', S. M. Feinberg,
Yu. M. Bulkin, I. Ya. Emel'yanov,
A. D. Zhirnov, A. K. Kalugin,
A. Ya. Kramerov, S. P. Kuznetsov,
E. P. Kunegin, K. K. Polushkin,
V. S. Romanenko, A. P. Sirotkin,
V. M. Fedulenko, V. A. Chebotarev,
and O. M. Glazkov

High-output uranium–graphite channel type reactors with thousands of process channels ("technological channels") have been in service in the USSR for many years. Power reactors of this type (Siberian nuclear power station [1]) have been in service since 1958 with operating temperatures in the neighborhood of 200°C and loop pressures up to 50 atm. The reactor of the World's First nuclear power station was started up in 1954 [2]. Two nuclear steam superheat reactors have been in operation at the I. V. Kurchatov nuclear power station at Belyi Yar (one since 1964, the other since 1967) at parameters as high as 520°C and 90 atm [3]. Vast experience has been accumulated to date in the operation of those systems.

The high degree of viability of these reactors, attributable to the possibilities of effecting monitoring and control channel by channel, is not a matter of record. These options have made it possible to detect malfunctioning in individual process channels well in time, so that these channels can be shut off and removed from service before malfunction and damage can spread on an extensive and dangerous scale throughout the entire system. Methods have been worked out and perfected for shutting off and repairing discrete components and subsystems, including the graphite stacking, to the point of replacing all of the graphite stacking in a major overhaul [4]. Such a comparatively vulnerable subsystem as the portion of the process channels located within the reactor core can be replaced during scheduled preventive maintenance operations, and even with the reactor still on power. Despite possible malfunctions in individual channels, then, the reactors have been operated successfully, on the whole, for decades of total operating time.

The development of large-scale high-output nuclear power cannot rest solely on experience in design, but must also be backed up by a large number of experimental design projects and related practical work. Even the performance of individual large-scale prototypes cannot constitute reliable evidence of the feasibility of developing this type of reactor for large-scale power production. Only generalization of many years of experience accumulated by industry can provide the reliable foundation needed. Consequently, the choice fell on the uranium–graphite channel type reactor as the vehicle for large-scale high-output nuclear power development, with due attention given not only to the experience accumulated but also to the progress contemplated in nuclear technology. It is of course of prime necessity that there be convincing arguments in support of the view that the trend being developed will at least not lag behind or turn out to be inferior to other reactor types in terms of costs, reliability, procurability of nuclear fuel, and so forth.

We may now enumerate the advantages of channel type reactors: (1) high reliability and viability of the entire system thanks to channel-by-channel monitoring capability, and the possibility of maintenance and repair of an individual process channel without unscheduled reactor shutdowns; (2) the possibility, in principle, of achieving reliable safety by subdividing the coolant circulating loop, with options of an autonomous process channel included; (3) virtually unlimited possibilities, in practice, of raising the power

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output on the basis of modular design components; (4) the possibility of building a steam generating unit without recourse to large-size pressure vessels, thereby expanding the production capabilities at the same time; 5) available option of organizing refueling operations with the reactor still on power; (6) perspectives of nuclear steam superheat; (7) flexibility of the fuel cycle, with the fuel cycle readily adaptable to fluctuations in the fuel market; e.g., the rate of consumption of natural uranium could be curtailed to a fraction, for example, by utilizing fuel of higher density or by using a mixed fuel charge (uranium + thorium), or heavy-water moderator.

The outstanding disadvantage of the channel type reactor is the multiple branching and comparative complexity of the circulation loop.

Some design improvements capable of greatly simplifying and contracting the circulation loop (see below), and contributing to making the steam generator unit and channel type reactor into as compact a package as the steam generator unit plus vessel type reactor, have been forthcoming.

The development of the traditional type of uranium–graphite reactors opens up new perspectives for nuclear power development, while the experience accumulated in this area provides a reliable foundation. This reactor type may also be used for other purposes, for instance in the production of energy and desalination of sea water [5].

The evolution of high-power uranium–graphite reactors is indissolubly linked with progress in nuclear engineering and in reactor materials technology. The various pathways of development open can be traced out against the example of the channel type uranium–graphite reactors, breaking them up into two groups to facilitate the analysis. The first group, centered about the achievement of high thermal efficiency, may be said to include the reactor installed in the World's First nuclear power station (1954). the reactors at the I. V. Kurchatov Belyi Yar nuclear power station (1963), the reactor with supercritical coolant parameters now in the design stages. The second group, centered about improvements in the fuel cycle, would then include the reactors at the Siberian nuclear power station (1958) and the RBM-K-1000 type reactors now being built. In the case of the reactors at the Belyi Yar nuclear power station, it was found possible to proceed to cooling of the tubular fuel elements using boiling water and superheated steam, even though steel-jacketed fuel elements were employed in those applications.

Relying on the current achievements of nuclear technology in the area of refractory zirconium alloys for pressure tubing and jackets of rod-type fuel elements with uranium oxide cores in water-cooled water-moderated pressure-vessel reactors [6], a decision was made to incorporate such design components in uranium–graphite reactors with the object of improving the fuel cycle over that characterized by the use of steel and tubular fuel elements (Belyi Yar power station), and improving the thermal efficiency over that characterized by the use of aluminum alloys (Siberian power station). The proposal to adopt that type of coolant was based on the successful operating experience accumulated in the management of boiling-water reactors with single-loop arrangements (Belyi Yar nuclear power station, VK-50 reactor).

It was in this setting that the RBM-K (boiling-water reactor of high power rating), a single-loop uranium–graphite reactor, appeared on the scene, using zirconium alloys as the basic structural material of the reactor core. The designs and materials selected in the construction of the first reactor units were closer to those verified in earlier practice, even though the possibility of considerable progress in nuclear reactor technology had been apparent from the very outset.

Construction of the first power unit of the single-loop nuclear power station (LAÉS), 70 km west of Leningrad and the largest such unit in the USSR and in Europe altogether, with a rating of 2 GW (el.) and two channel type uranium–graphite (RBM-K-1000) reactors cooled by boiling water (Fig. 1), is now nearing completion. This power station is the pacesetter of a new line of nuclear power stations of its type now under construction, constituting the future base for production of a major portion of electric power to be fed into the national grid in the European sector of the Soviet Union.

LAÉS Power Station Layout and Basic Parameters

The main kernel of the nuclear power station consists of two power units rated at 1 GW (el.), sharing a common machinery room and separate reactor bays, systems for conveying fuel, control panels, and a common room for gas cleanup and purification of the primary-loop water. Each power unit incorporates a RBM-K-1000 reactor with a circulation loop and auxiliary systems, steam lines and condensate-feedwater lines, and two K-500-65 turbogenerators rated at 0.5 GW (el.) each.

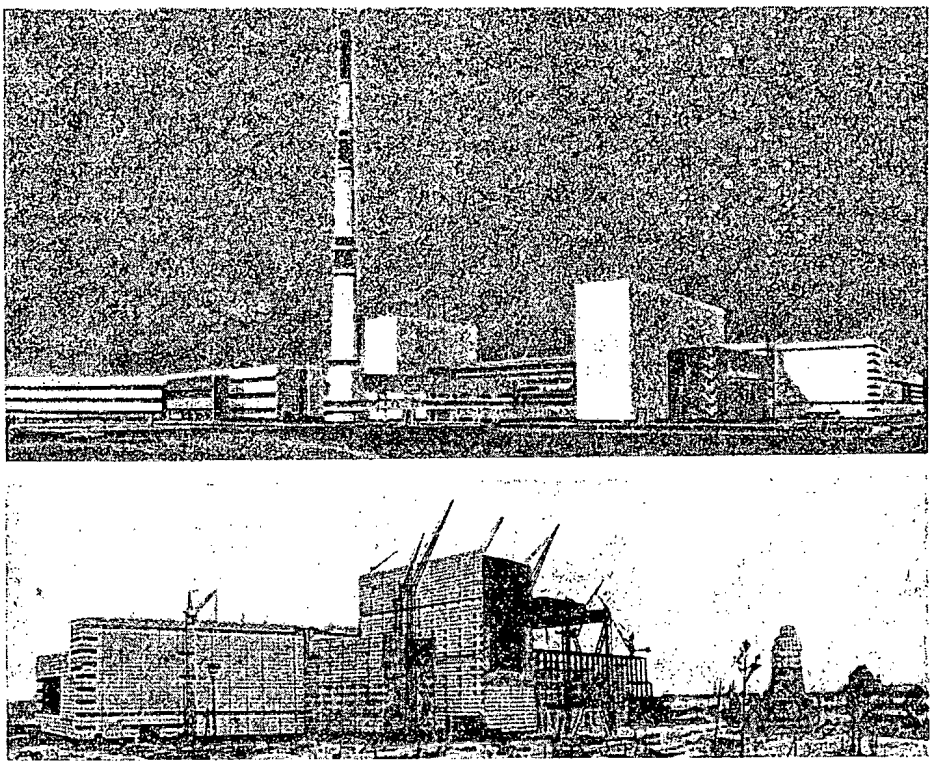

Fig. 1. General view and panoramic view of the construction work on the LAÉS nuclear power station near Leningrad.

The circulation loop consists of two parallel subloops, each subloop including two separator drums, four circulation pumps with a set of valves and piping of medium diameter (D_y = 300) and large diameter (D_y = 800, 1000), and 22 distributor groups of headers (D_y = 300) supplying the reactor channels. There is a common system for ion exchange resin cleanup of the loop water, with flowrates amounting to 4% of the reactor steam rating.

Saturated steam under 70 atm pressure leaves the separators through eight steam lines (400 mm diameter) to reach the two K-500-65 turbines. The condensate is returned via the steam trap and five low-pressure reheaters as recycle to the deaerator, and on to the feedwater pumps via valves controlling the fill level in the separators.

The turbogenerators are connected as blocks to an open-air electric power substation rated at 330 kV. The power for the local needs of the LAÉS power station comes from the generators. Standby power is provided by switching in supplies from 110 and 330 kV power lines through standby transformers, and in the event of a total power blackout power can be obtained from a separator generator in a hydroelectric power station and from diesel generators which switch on automatically in three minutes' time. Power consuming units which cannot tolerate any interruption in their power supplies are connected to a bank of storage batteries. Excess steam appearing when the shutoff valve of one or both turbines is closed is vented through pressure relief units to the turbine condensers, and if the vacuum there has been disrupted the excess steam is routed to the steam receiver unit (bubblers and process condenser) which also collects discharge from the safety valves, so that steam is not vented to the atmosphere.

The power rating of a single LAÉS power unit is 3.2 GW (th.) and 1 GW (el.) plus a bleed of 70 Gcal/h heat for local space heating. The saturated steam pressure upstream of the turbine is 65 atm when the feedwater temperature is 165°C. The steam rate is 5800 tons/h. The operating conditions are: base load with possible shift to control mode; water treatment neutral with no correctives. The basic structural materials are: circulation loop austenitic steel and zirconium; steam lines and condensate lines carbon steel; condenser turbing MNZh alloy; tubing of preheaters stainless steel; metallic structures and enclosure of reactor carbon steel. The fuel is uranium dioxide enriched to 1.8% (stationary conditions), the average burnup 18,500 MW-day/ton. The weight of the stationary fuel charge is 180 tons uranium; the first charge is 180 tons uranium 1.1% enriched or 155 tons uranium 1.8% enriched.

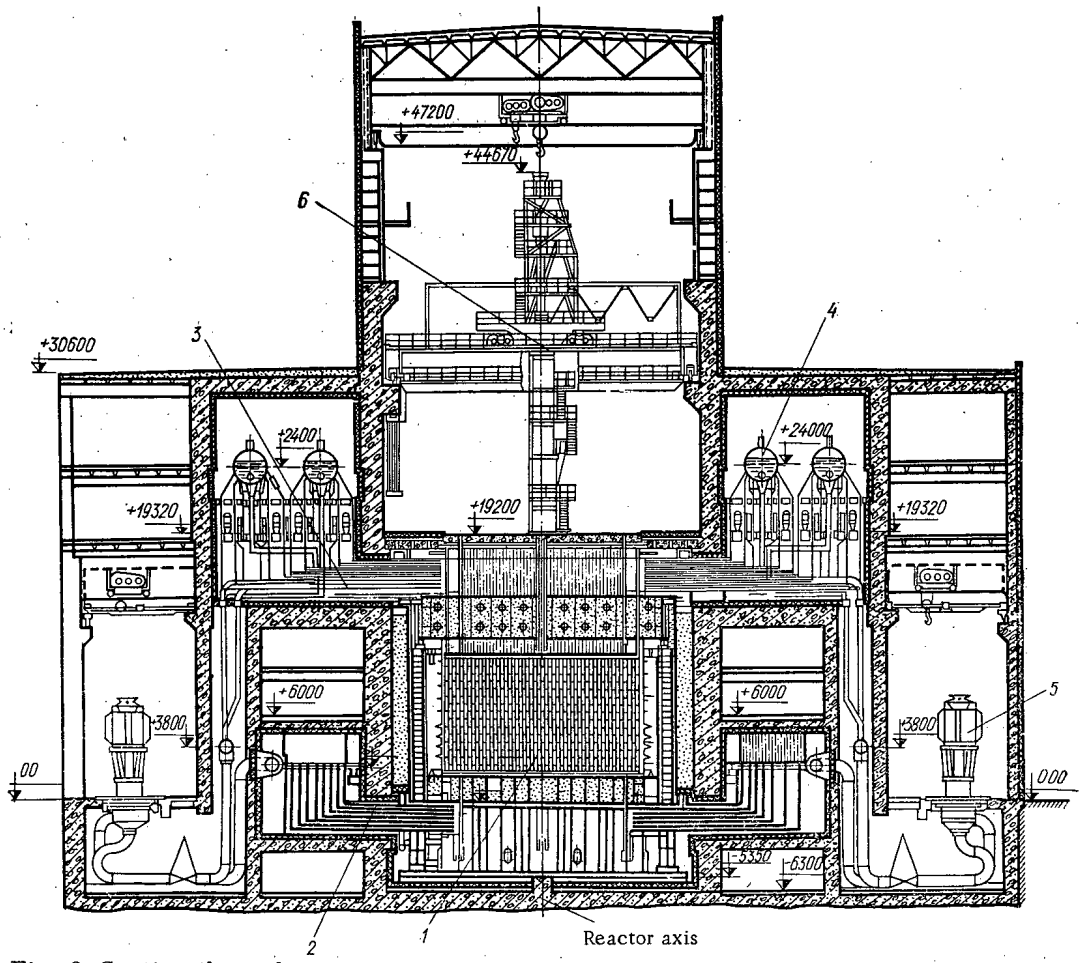

Fig. 2. Section through reactor and multiple circulation loop: 1) core; 2) water supply lines; 3) steam–water mixture bleed lines; 4) steam separator; 5) pump; 6) refueling machine.

The characteristics of the basic process equipment (in the power unit) are: separation gravitational in horizontal stainless steel clad separator drum; drum diameter 2.3 m, drum length ~30 m, drum weight 200 tons, four drum separators in all; eight pumps, centrifugal electrically powered pumps with sealed pump shaft outlet and flywheel, throughout 7000 m^3/h at head of 200 m H_2O and 1000 rpm, weight including 5,500 kW electric motor ~100 tons; two saturated steam turbines type K-500-65, single-shaft, double-flow (one high-pressure cycle and four low-pressure cycles), turbine length 39 m, shaft speed 3000 rpm, weight 1200 tons; steam separator and steam reheater both located between high-pressure cycles and low-pressure cycle; two 500 MW (el.) generators, three-phase, 50 cycles/sec with hydrogen cooling and water cooling.

Four channel systems are used in the RBM-K system, with the possibility of setting up a refueling system operable with the reactor on power. This makes it possible to raise the load factor of the nuclear power station and lower unproductivity losses in the absorbers of the protection and control (rod) system, and also to minimize vapor discharges through the possibility of detecting severe failures or leaks in fuel assemblies and replacing these faulty assemblies promptly. The weight of the refueling machine, shielding included, is ~465 tons.

Reactor Design

The reactor is located in a concrete pit 21.6 × 21.6 × 25.5 m. The weight of the reactor is transmitted to the concrete via welded metallic structures which simultaneously perform the function of shielding and form, together with the enclosure envelope, a cavity filled with a helium–nitrogen mixture constituting the reactor space within which the graphite stacking is accommodated (Fig. 2). The graphite stacking consists of graphite blocks with cylindrical holes assembled in columns. The columns are joined by cooled beams radiating outward in holes in the peripheral columns.

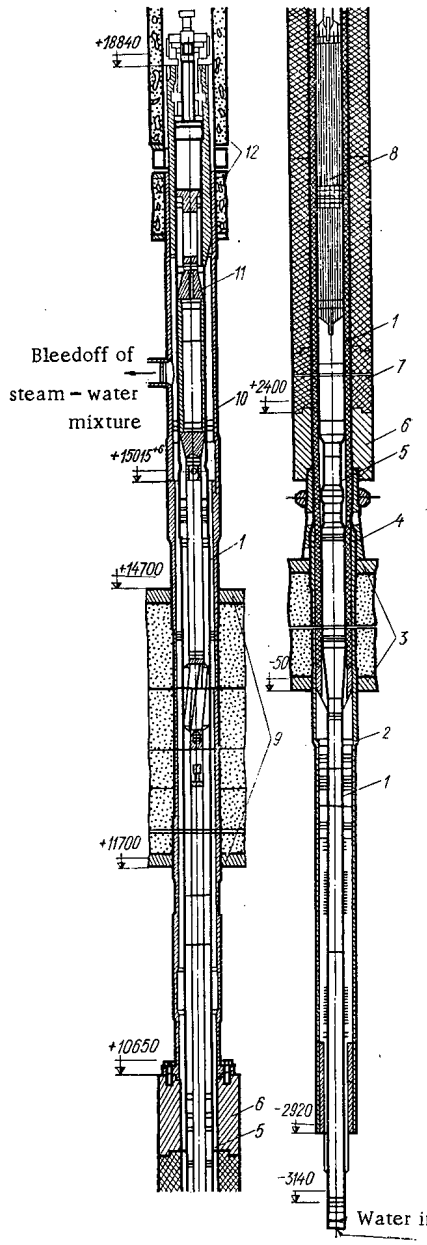

Fig. 3. Fuel channel: pressurized tube; 2) channel passage; 3) bottom shielding plate; 4) support sleeve; 5) adapter (steel–zirconium); 6) thermal shield; 7) stacking block; 8) fuel assembly; 9) top shielding plate; 10) fuel channel head; 11) fuel assembly suspension support; 12) removable shielding plug.

The fuel channels (about 1700 of them) are located in tubular passages which are welded to the top and bottom metal grids of the reactor (Fig. 3). The top and bottom parts of the channels are made of stainless steel, while the central tube sized 88 × 4 mm is made of Zr + 2.5% Nb alloy exhibiting quite satisfactory mechanical and corrosion-resistant properties. The zirconium part of the channel is joined to the steel parts by means of special welded steel–zirconium adapters (Fig. 4) which are subjected to tests to determine their expected length of service life. The channel accommodates one fuel cluster with two assemblies. Each fuel assembly comprises a set of 18 fuel elements, with the fuel meat section extending 3.5 m in length. The fuel element consists of a tube 13.5 × 0.9 mm in diameter, made of zirconium alloy with uranium dioxide pellets. The water flowrate through the channels is adjusted twice during the reactor campaign to match changes in the fuel channel power output, by adjusting control valves installed on the process piping. The crucial problem is how to extract heat from the graphite stacking. An arrangement in which the stacking is cooled by heat transfer to pressurized tubes (Fig. 5) was decided upon with the object of simplifying the reactor design.

The protection and control system is designed to control reactor power drops at a rate of 4%/sec, and acts to keep power performance constant in the event of accidents or malfunctions not requiring shutdown of the reactor. When necessary, the reactor can be shut down completely at a power loss rate of 8%/sec. The protection and control system makes use of highly reliable semiconductor and noncontacting components. Standby equipment is built into the system.

The power release monitoring system that monitors the amount of power liberated with respect to reactor height and reactor radius used neutron-sensing components installed in the parts of the fuel channels and channels with measuring and sensing components for the protection and control system.

The soundness of the fuel elements is monitored, and leaks detected, by measuring the activity of the steam–water mixture in the piping at the entrance to the separators, with the aid of paired scintillation type γ-ray spectrometric sensors mounted on moving platforms and periodically monitoring the piping. The water flowrate through the channels is also monitored by flowmeters installed at the inlet to each fuel channel and at the inlet to the protection and control channel. The soundness of the channel tubes is monitored by checking the changes in the property of the gas flowing outside the channel.

The vapor mixture is circulated through a closed loop. The vapor mixture is purified of water vapor and products of graphite oxidation in the loop, and the prespecified composition of the mixture is also monitored and maintained at its proper levels.

Some of the reactor system components were developed, in the course of designing the reactor, in approaches similar to those used in the design of components which have made a good name for themselves over an extended period of operating history with reactors of this type.

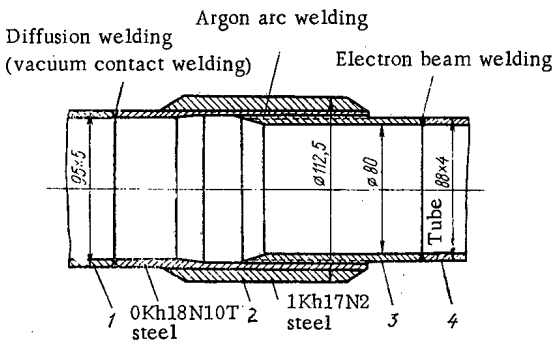

Fig. 4

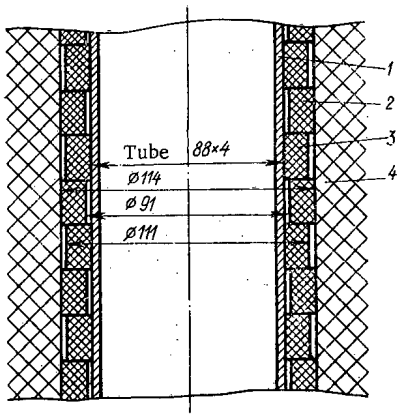

Fig. 5

Fig. 4. Joining of zirconium alloy tube to stainless steel tube: 1) tube of 0Kh18N10T steel; 2) union of bimetal sandwich; 3) nipple adapter, Zr + 2.5% Nb alloy; 4) tube of Zr + 2.5% Nb alloy.

Fig. 5. Placement of process channel in graphite stacking: Zr + 2.5% Nb alloy tube; 2,3) outer and inner graphite annuli, respectively; 4) graphite stack.

A program of experimental and investigative work and reactor tests, including heat-transfer and service-life tests, were carried out on the fuel channels and models of fuel assemblies on special test stands, and subloop reactor tests were carried out on a full-sized fuel assembly, in order to confirm the performance qualities of the redesigned components and subsystems.

Monitoring, Control, and Adjustment of Power Station

One of the major problems in the building of the LAÉS power station was how to avoid shutdowns of the power station except in extreme situations where that recourse is absolutely unavoidable, and, still more to the point, eliminating any chance of reactor shutdown in response to spurious emergency signals.

The refueling system and the system for monitoring leaks, operable with the reactor on power, are designed directly or indirectly to facilitate that purpose, as is the elimination of the branched system of impulse pressurized water tubes for large-scale channel-by-channel monitoring, minimization of shielding on the process equipment where easily triggered protection devices would cause reactor power unit shutdown, so that the protection and control equipment will be activated only when a signal is present from not less than two (or three) channels carrying measurements of reactor system parameters.

System scramming which would immediately bring the reactor to subcriticality even when an emergency signal is not present is designed for only rare eventualities. Instead, it was decided to rely upon specialized shielding measures (specialized in terms of specific groups of possible accidents and malfunctions) allowing for a controlled drop in power at a sufficiently fast rate and to a level which would guarantee extraction of the heat in accidents and malfunctions belonging to that specific group. If the signal triggering the protective action disappears, then the power drop is discontinued and normal functioning is resumed.

The radiation environment is sensed by the same radiation sources as are used in pressure vessel type boiling-water reactors (instruments sensing O^{16} in the steam lines, F^{18} and corrosion products in the condensate stream, O^{13} and RBG at the discharge end of the condensate ejector, etc.). The activity of vapor discharges from the condensate ejector is expected to be moderate (~3000 Ci/day) even in the event of a large number of failed fuel elements, thanks to the system for process monitoring and refueling with reactor on power, and the systems incorporating combustion of explosive mixtures and sorptive retention of RBG on activated charcoal.

Only those fuel assemblies containing fuel elements with marked symptoms of leakage will be removed and replaced ahead of schedule. Even where a slight leakage is detected, there will be no need to shut down the reactor, and it will suffice simply to extract the fuel assemblies from the faulty fuel channel and use the refueling machine to set a specially designed plug in that channel.

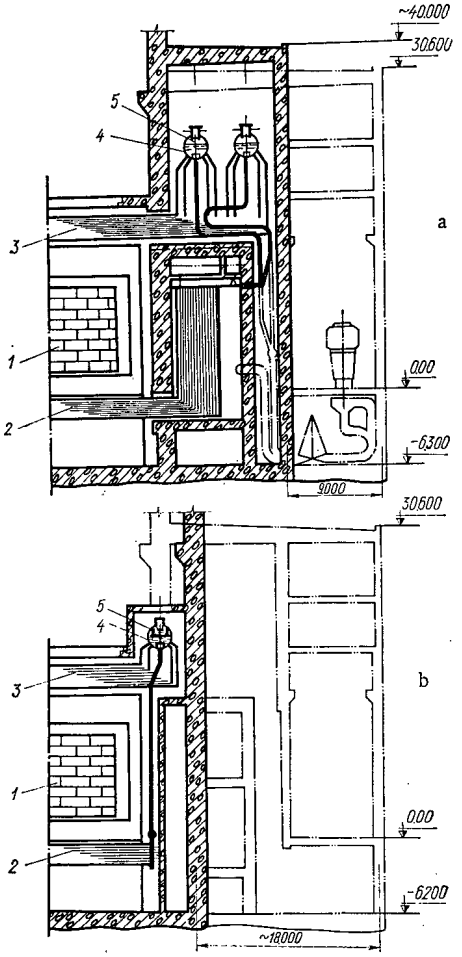

Fig. 6

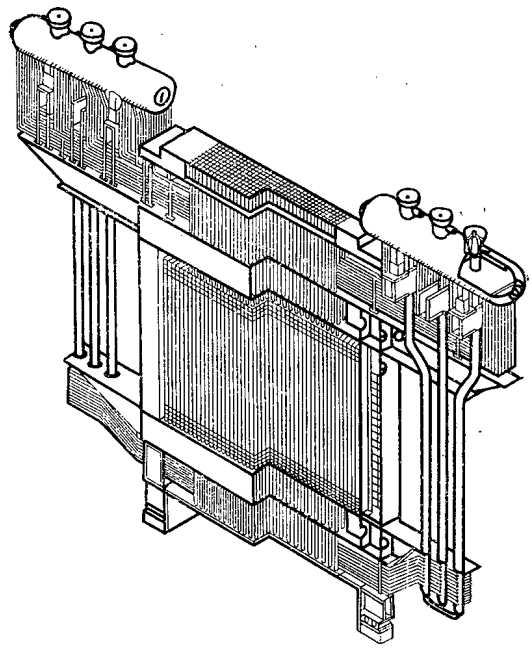

Fig. 7

Fig. 6. Variants in arrangement of components (large pumps and headers eliminated): a) natural-flow separation; b) forced-flow separation (LAÉS power plant variant indicated by dot-dash lines); 1) core; 2) water supply lines; 3) steam–water mixture extraction lines; 4) steam separator; 5) pump.

Fig. 7. Section of modular variant of RBM-K type reactor.

Control of power station performance is designed for base load and a followup mode by maintaining the pressure upstream of the turbine at a set point through action of the controller on a throttle valve (in operation at base load) or on the reactor power controller (in the followup mode). Operational up-to-the-minute monitoring and control of the power facilities is handled by computer and data-processing machinery.

Nuclear Safety

Some features guaranteeing radiation safety are built into the reactor system design:

1) high-reliability protection and control system including about 180 independently acting absorbers combined with groups with autonomous sensors, cables, comparison equipment, equipment for amplifying signals, and power supplies;

2) equipment for emergency heat removal (flywheels on the principal loop pumps, standby power sources for local needs, routing of feedwater to a discharge header, etc.), eliminating any large-scale damage to the cladding of the fuel elements due to any or all of the indicated causes of accidents or malfunctions, including total power blackout, outage of two turbines simultaneously, leaks in piping of 300-400 mm diameter, etc. Even complete rupture of piping of that size outside the grounds of the LAÉS power station proper can be handled safely through these measures;

3) equipment for making periodic checks on the state and functioning of all the subsystems and systems responsible for maintaining radiation safety, including periodic checkouts of the state of large vessels and headers, thereby practically eliminating any instantaneous total rupture of the equipment or total breakdown;

4) plenum chambers and surge tanks for steam, to eliminate large-scale venting of steam to the atmosphere in emergencies.

It is also important to note that there will be no dangerous positive reactivity surge in response to any kind of damage, accident, or failure affecting the circulation loop.

Generalization of the operating experience acquired with conventional pressure-vessel machinery and uranium–graphite channel type reactors already in operation in the USSR shows that a sudden instantaneous failure or burst of large-diameter piping and of drums designed to specifications and fabricated in conformity with accepted technology of high-pressure unfired vessels is extremely unlikely at the moderate pressures and temperatures involved, and given the intensity and quality of the inspection and monitoring processes.

Leaks in the primary loop have been observed. But these leaks developed at a comparatively slow rate and in no instance led to an instantaneous rupture of large-diameter piping, even though breaks in smaller piping 1-2 cm in diameter often result from corrosion and vibration effects. To cope with this, measures were taken to institute emergency cooling of the core and to eliminate any damage to fuel elements or release of fission products into the power station rooms or into the surroundings. With the present siting of nuclear reactor power generating stations near large cities and the corresponding public-health isolation zone around the plants, the reactors are for practical purposes completely safe with regard to the populations of the nearby localities, as is clearly evident when examining the case of the Belyi Yar nuclear power station which has been in operation and generating power since 1964 [7].

Improvements in Circulation Loop and Enhanced Nuclear Safety

Future plans for building a large number of reactors that necessarily have to be situated closer to large population centers, and where the public-health isolation zone around the nuclear plant will have to be contracted, the problem of further improvements in the level of nuclear safety typifying nuclear power generating stations comes up on the agenda.

Analysis of the consequences of failures of 300 mm diameter pipe has demonstrated that a large volume of water and steam is discharged when the water volume of the loop is combined, and that it is difficult to localize this discharge without severe additional losses. But it is precisely the channel reactor design which opens up the possibility, in principle, of finding a successful solution to this safety problem. Here there is no need to combine the water volume of the entire reactor as is the case in pressure-vessel type reactors. On the contrary, the analysis shows that the small number of high-power pumps can be reduced by a large number of low-rating pumps, that large headers and branches in the loop can be eliminated by relying instead on a large number of separate sections which are sufficiently autonomous in their operation. The resulting complication of the control system is not that great, but a failure in the loop is in any case localized within the confines of one section of the loop. The existence of communications between different sections of the loop in pairs does not hinder such localization of damage and failure, so that these steam links in the loop can be retained without worry.

Axial-flow turboseparators with a high steam load per unit volume, and jet pumps with a comparatively low throughput, are widely used in the pressure-vessel type boiling-water reactors.

Fragmentation of the circulation loop brings up the possibility of complete malfunctioning of the valves, headers, and large-diameter pipes, while the use of turboseparators also makes it possible to situate the latter in vessels of moderate diameter. The comparatively small amount of water present in each section of the loop, possibly one-tenth or even one-hundredth the amount present in conventional loops, or less, facilitates accommodation and localization of steam and water in the event of any disruption of the loop.

In addition to the points made above, the fragmentation of the loop and the use of small-size separators also makes it possible to locate these machines closer to the reactor, thereby reducing the length of

TABLE 1. Basic Characteristics of RBM-K Type Reactors

Parameters	RBM-K-1000	RBM-K-2000	RBM-KP-2000	Reactor supercritical parameters [8]
Power output, MW:				
electrical	1 000	2 000	2 000	1 000
thermal	3 200	6 280	5 620	2 280
Steam parameters upstream of turbine:				
pressure, atm	65	65	65	240
temperature, °C	280	280	450	535
Number of fuel channels	1 693	1 404	1 404	1 022
Uranium loading, tons	180	320	294	52
Average fuel burnup, MW-day/ton	18 500	28 000	24 000	40 000
Fuel	Dioxide	Dioxide	Dioxide	Carbide
Peak channel output, kW	3 000	6 580	Evap Ch 6350 Steam Superh 4500	Steam Superh 2200
Peak heat flux, 10^6 kcal/m^2·h	0,7	0,9	0,9	2,5
Fuel irradiation level, MW/ton	17,8	19,6	19,2	44
Fuel-element cladding material	Zirconium	Zirconium	Zirconium - steel	Steel

the outgoing steam piping runs by a factor of three or four. This also results in improved interchannel stability in circulation, since stability varies in inverse proportion to the steam volume, and since it is the vapor tension of the steam void that is responsible for the development of pulsations in flowrate. This design solution based on fragmentation of the loop with the use of turboseparators and closer siting of these turboseparators to the reactor is a way of contracting the volume of the loop, contracting the water volume of the loop, and cutting down on the amount of metal required in that part of the plant by a factor of two or three.

Of course, the large pumps have to be replaced by a large number of smaller pumps of suitable type. Feedwater pumps in such variants as steam-jet heat pumps and hydraulically driven pumps are convenient only when the multiplicity of recirculation is small, and electrically driven pumps with constant shift rpm lead to annoyingly rapid drops in steam content when the power falls off, so that a large reserve supply of water is required under the separator level, which in turn means that there would be no futher gain in switching to small-size high-intensity separators such as turbo-steam separators. The use of steam-driven turbopumps incorporated into the loop in the form of a vertical single-shaft unit, with the pump impeller located below floor level and the turbine drive located above floor level, favors more compactness and improved safety in the operation of the circulation loop. Those pumps have no shaft outlet, are much more compact and much lighter than electrically driven pumps, and exhibit the important properties of proportionality of rpm and pump throughput to the flowrate of steam from the reactor. That is what allows us to get by without positioning the pump in a very deep shaft, while curtailing the water reserve in the separator several times, and also allows us to resort to emergency cooldown automatically, through the agency of ponderous flywheel masses (Fig. 6).

A basically new system for extracting heat from the fuel channels has been worked out more recently, one which opens up the possibility of further radical simplifications in the circulation loop. As the weight flowrate of the two-phase coolant decreases, the critical steam content and the critical burnout loads increase. A "multitiered" arrangement of organization of heat removal in fuel channels, with the fuel channels dismembered along their height into several "tiers" communicating with the water feed lines and steam extraction lines located within the pressurized tubes in a parallel arrangement, has been proposed. The weight flowrate of coolant in each such "tier" would be reduced by as many times as the number of "tiers" the fuel channel is broken up into, provided there is no change in the total flow area presented by the cross section of the assembly of fuel elements.

The rise in steam content (at the operating point) to 30%, and even to 50%, cuts down the number of cycles of circulation through the loop that is required by a factor of two to three, and the hydraulic resistance presented by the fuel channels decreases to roughly 5-6 atm. Since the power used up in circulating the coolant is proportional to the product of the circulation multiplicity factor by the hydraulic resistance, the power drops so low that the use of jet pumps with head supplied by the feedwater pumps becomes an effective procedure. The principal radioactive circulation loop of the reactor is thereby freed from any need of mechanical pumps, which means simpler operation and cheaper operating costs.

Moreover, the amount of water in the reactor core is also cut back considerably, and is largely replaced by steam. The depth of fuel burnup increases and the physical stability of the plant is improved, since the operating point becomes displaced toward the region of higher steam void content. The amount of metal used in the design of the circulation loop, and the volume of room space taken up by the circulation loop, are both cut back considerably.

Uranium – Graphite Reactors being Developed

The materialization of some concepts on improvements in RBM-K-1000 type reactors found reflection in the designs of the RBM-K-2000 reactors and the RBM-KP-2000 steam superheat reactors rated at 2 GW (el.). In contrast to the RBM-K-1000 type, these reactor types were designed with scaled-up channels. The evaporation channels in these reactor types are designed almost identically, while the fuel elements are identical with those in the RBM-K-1000. The design of the reactor cores differs for the most part only in the presence of the steam superheat channels in the RBM-KP-2000 reactor design. That also accounts for the difference in the process flowsheets. The principal characteristics of the RBM-K-2000 reactor type are listed here in Table 1. The nuclear power plant flowsheet is basically similar to the flowsheet worked out for the RBM-K-1000 reactor power plant.

Within the confines of the reactor core, the evaporation channel is a stepped affair, made of zirconium tubes, with the large-diameter tube located above the center of the core. The expansion of the diameter of the zirconium tube in the zone of intense steam generation allows for placement of intensifiers, improvements in the conditions governing cooling of the top of the fuel assemblies, increasing reserve margins up to critical burnout levels, and also greatly reducing the hydraulic resistance presented by the channel as a whole.

The nuclear steam superheat channel features two-way flow of coolant. The saturated steam flowing downward through the outer annular clearance cools the channel tube, so that it can be fabricated of zirconium alloy. Steam superheat is achieved in the inner telescoping portion of the channel, where the fuel assemblies are positioned. The fuel assembly comprises a tube of stainless steel sized 10 × 0.3 mm, filled with uranium dioxide pellets.

The channels are accommodated in a square grid with a pitch of 320 mm. The total number of channels is 1404, of which 1050 are evaporation channels (Evap Ch in Table 1) and 354 are steam superheat channels (Steam Superh in Table 1). Forced-flow cooling of the stacking by circulating nitrogen is planned with increased pitch between the channels and increased specific power output in the RBM-KP-2000 generation of reactors. The blowers and heat exchangers of the gas cooling stream are built directly into the reactor enclosure.

Coolant circulation through the evaporation channel and steam superheat channels is handled by the turbopumps installed in the drum-separators. The protection and control system of the RBM-KP-2000 reactor, in contrast to that of the RBM-K-1000, uses control rods situated outside the grid of fuel channels. The absorbing part of the rod is made in the form of slugs of boron carbide in a graphite matrix.

The rods move on special paths under the action of push rods inserted below the reactor. Cooling is by gas circulating through the cooling system of the reactor stacking.

Efforts to maximize the efficiency of the nuclear power station led to the completed design of a reactor adapted to supercritical coolant parameters. Some of the design characteristics of the 1000 MW (el.) uranium–graphite reactor producing steam of supercritical parameters are included in Table 1.

In evaluating the cost indices of the nuclear power stations based on channel type reactors, we can see clearly that these power stations are not inferior in cost performance to nuclear power stations based on reactors of other types, such as gas-cooled graphite-moderated reactors, or water-cooled water-moderated reactors.

Large-scale nuclear power development brings to the fore the problem of organizing industrial methods of fabrication and assembly of the reactor. Efforts to increase reactor unit power output pose the problem of working out a reactor design such that reactors of increasingly higher output ratings can be built with minimum changes and without radical restructuring of the production facilities and plant, i.e., on the basis of unitized and standardized modules and structural designs. The possibilities inherent in channel

type uranium–graphite reactors make it possible to find solutions to this problem. The first step taken in that direction is the design plan for a sectionalized modular uranium–graphite channel type reactor.

The reactor is assembled from standard sections: central and edge sections joined according to a prespecified plan at the rigging and erection site. The core forms a rectangle in plan view. The length of the rectangle is determined by the number of central reactor sections (Fig. 7). A reactor of practically any power rating can be built from those sections simply by increasing the number of central sections. The sections function as separate component units of the reactor system, and incorporate the necessary equipment plus control and monitoring hardware, and consist of individual transportable modules. The width of the section is determined by the diameter of the separator.

Some Possibilities Inherent in the Fuel Cycle

One of the principal features and advantages of channel type reactors is the broad range of application of the different fuel cycles. The possibility of refueling with the reactor on power, channel-by-channel process monitoring, and the comparatively low multiplication factor, combine to open up new pathways for flexible control of the fuel cycle.

Depending on the future conjuncture regarding procurement and mining of natural uranium, e.g., in case the price of natural uranium rises, conversion to utilization of fuel elements with denser (in uranium) meat cores combined with comparatively low burnup is an option still open. This makes it possible to lower needs for procuring natural uranium by increasing the U^{238} burnup ratio, and also by using uranium of about 1% enrichment, which may be economically feasible. Plutonium breeding is another possibility, and may gain in importance as the power output of fast reactors increases. Without necessitating a change in design, it would also be possible to effect a transition to thorium composition with U^{233} or U^{235}. The fuel cycle can be improved by lowering the operating pressure (which would make it possible to thin out the walls of the channel tubes and the cladding of the fuel elements) and by cutting down on the amount of water present in the process channels by increasing the steam volume.

Finally, if even greater savings in natural uranium resources are necessary, it would be possible to replace all or much of the graphite moderator by heavy-water moderator, while retaining the basic design and process flowsheet features of the reactor.

Curtailment of the volume of water and structural materials in the core by lowering the pressure and by improving the technology of refractory zirconium alloys will contribute to building up reserves, over a long period, for gradual improvements in the fuel cycle and corresponding cutback of the fuel component in electric power generation costs, it should also be pointed out.

The channel type reactors now being developed are thus seen to be flexible and adaptable to possible changes, even difficult-to-predict changes, in the outlook for natural uranium procurement, and plutonium breeding demands. This means that we can consider the present developmental outlook of channel type reactors to hold good for a protracted period of time ahead.

The data cited above, and the possibilities of improving the reactor design and fuel cycle, allow us to express confidence that nuclear power generating stations of the Leningrad type now being built will be, to begin with, economically competitive with power stations burning fossil fuels anywhere on the territory of the European part of the USSR (not to speak of the appreciable by-product of advantages to the national economy in keeping the atmosphere free from pollution by combustion products, and the considerable savings, in cost and in available means of transportation, in minimizing long-haul shipments of fuel), and secondly these reactors are at the very least not inferior to other basic types of power reactors (and sometimes even surpass the latter in performance).

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DESIGN AND OPERATING EXPERIENCE WITH FAST REACTORS IN THE USSR

A. I. Leipunskii, O. D. Kazachkovskii,
F. M. Mitenkov, V. V. Orlov,
V. A. Stekol'nikov, and V. I. Shiryaev

The considerable experience accumulated in the USSR in more than 20 years of work on sodium-cooled fast reactors has allowed us to proceed to the construction of the first atomic power plants with the BN-350 and BN-600 reactors. The design and construction of these reactors required the creation of a broad experimental base for the physical, heat-engineering, technological, material properties, and other studies, and for tests of reactor equipment. The experiments were directed particularly toward the BR-5 and BOR experimental fast reactors.

We give a status report on the operation and construction of fast industrial reactors in our country with particular attention to experimental tests of equipment for fast-reactor power plants under construction. As a result of the start-up and operation of the BN-350 and then the BN-600 we shall accumulate industrial experience with reactors of these two types in the next few years. This experience will permit us to proceed, even in this decade, to a more extensive construction of fast-reactor power plants as the result of the industrial assimilation of BN-600 equipment. We hope these plants will generate electrical power at an economic advantage.

The future prospects of fast reactors in our country are bound up with high-power reactors – 1000 MW or more. The structural development of such reactors, based on principles established with the first reactors, confirms the possibility of constructing fast reactors of such power. The accumulation of design and operating experience points the way to an appreciable improvement of the technical and economic characteristics and the reliability of fast reactors. However, the practical realization of high-power reactors with these improvements, planned for the second half of this decade, must follow the construction and mastery of the first fast-reactor industrial power plants and the accumulation of operating experience with them.

In this paper we do not discuss fuel problems; these are considered in other papers of the present conference.

BR-5 Reactor

January 1971 marked the twelfth anniversary of the start-up of the BR-5 reactor. The principal operating characteristics of the reactor during the years 1959-1970 are listed below:

Total time of operation of reactor at power	55,000 h
Including time at nominal level	23,000 h
Actual number of days in operation	1,426 h
Including those with:	
PuO_2 fuel elements	468
UC fuel elements	958
Time of circulation of sodium in primary circuit	70,000 h
Including time at 450-510°C	28,000 h
Primary circuit filtration system operating time	30,000 h

State Committee on the Use of Atomic Energy in the USSR. Translated from Atomnaya Énergiya, Vol. 31, No. 4, pp. 344-352, October, 1971.

Number of pump repairs	
in primary circuit	30
in secondary circuit	37
Number of replacements of cold traps	
in primary circuit	9
in secondary circuit	2
Number of complete core refuelings	4
Number of planned additional core loadings	35
Maximum integrated neutron flux	
wall of loop channel	$7 \cdot 10^{22}$ neutrons/cm^2
central reactor tube	$4 \cdot 10^{22}$ neutrons/cm^2
uranium monocarbide fuel elements	$4 \cdot 10^{22}$ neutrons/cm^2
plutonium dioxide fuel elements	$3 \cdot 10^{22}$ neutrons/cm^2
Maximum fuel burnup	
in uranium monocarbide fuel elements	5.9%
in plutonium dioxide fuel elements	6.7%

From 1959 to 1964 the reactor operated with plutonium oxide fuel, and in 1965 the reactor core was loaded with uranium monocarbide fuel elements. Burnup in most of the plutonium dioxide fuel elements reached 5-6.7%, and in uranium monocarbide up to 4-5.9%. It was established that the 0Kh18N10T steel cladding on most of the monocarbide fuel elements was ruptured in a burnup of 4.0-4.6%, whereas all the monocabide elements clad with Kh16N15M3B steel remained in working order for burnups up to 5.5-5.9%. The plutonium dioxide fuel element failures resulted from fuel swelling and the loss of plasticity of the steel cladding. The failures of the monocarbide fuel element cladding were due to the deterioration of heat transfer from elements deformed in a nonuniform temperature distribution, and to the inferior long-term strength characteristics of 0Kh18N10T steel in comparison with Kh16N15M3B.

From 1967 to 1969 eight experimental absorbing elements (EAE) containing boron compounds (B_4C, CrB_2, EuB_6, etc.) and Ta, including unsealed elements, were irradiated in the reactor core. The elements received a neutron dose of $3 \cdot 10^{21}$ neutrons/cm^2. An examination of the EAE in the hot laboratory showed them to be in good operating condition.

For a considerable time the reactor operated with some leaking fuel elements which allowed fission products to enter the coolant and gas spaces of the primary circuit. Under these conditions particular attention was paid to monitoring the fission product content of the primary circuit and the integrity of the cladding of the fuel elements unloaded from the reactor. Systems were constructed and methods developed for monitoring the delayed neutron and fission product activity in the core coolant.

Radioactive fission products were removed from the sodium of the primary circuit periodically by a cold oxide trap which collected iodine and cesium well. A system for purging the argon in the primary circuit of gaseous radioactive fission products by using a carbon adsorber was also constructed and tested. After several hours of purging, the level of activity is lowered to the equilibrium value, when the removal of xenon isotopes is compensated by their influx from the sodium. In the BR-5 reactor, where 70 ± 5% of the xenon is in the sodium [1], the xenon activity in the argon is lowered by a factor of 10 during the first stage of purging; subsequently the sodium is purged of xenon. The high radioactivity of the sodium vapor traps in the gas scrubbing system should be noted. It was established that the Cs^{137} content in sodium condensed from vapor is 20 times higher than in the primary circuit coolant.

From 1969 the primary circuit coolant was monitored regularly for the content of nonmetallic and metallic contaminants. Seven sodium samples removed from the circuit during the last two years showed a high carbon content [$(0.45\text{-}1.7) \cdot 10^{-2}$%]. A possible source of entrance of carbon into the circuit is the bearing lubricant of the circulating pumps. The nitrogen content in the sodium is $(1\text{-}3) \cdot 10^{-3}$% and the hydrogen content is $(0.4\text{-}2) \cdot 10^{-3}$%.

The operation of the BR-5 on a monocarbide core was concluded in May of 1971. During 1971-1972 it is planned to make a series of changes in the BR-5 to increase its power to 10 MW. In addition to doubling the reactor power it is proposed to insert a third reactor loading of plutonium oxide fuel with an attainable burnup of 10% and a power density up to 780 kW/liter. The BR-5 will be forced by increasing the heating of the coolant in the reactor while maintaining the average sodium temperature at the core outlet at 500°C,

which will speed up the experiments on the irradiation of fuel and materials in the reactor by a factor of three of four.

BOR Reactor

The fast research reactor BOR was started up at Melekess in the USSR at the end of 1969 at a power of 60 MW [2]. The start-up of this reactor significantly extends our capabilities for experimental physics and engineering research on sodium-cooled fast reactors. Having high maximum parameters (power density up to 1100 kW/liter, sodium coolant temperature up to 600°C) the BOR reactor permits large-scale experiments on different fuel materials and compositions to a high burnup, investigations of absorbing elements (SUZ) of various materials and constructions, and studies of structural materials irradiated by large integrated neutron fluxes. The BOR reactor also permits experimental tests and development of systems and equipment being designed for fast power reactors.

The BOR reactor was built in a short period; design was begun in 1963 and the reactor was put into operation in 1969. During that time considerable experience was accumulated on design, construction of equipment, assembly, and start-up of the reactor [3]. All the main reactor units (sealing of the rotating plugs, reactor vessel connections, pressure chamber, drives and SUZ mechanisms, fuel assemblies etc.), the basic equipment (pump, steam generator, intermediate heat exchangers, accessories, etc.), instruments (level meters, monometers, thermocouples) were tested on models and prototypes. Before the reactor was put together an assembly check was made on the reactor vessel, the rotating plugs, the header, the fuel assembly mock-ups, the SUZ drives, etc. Careful attention was paid to the assembly process and high technological and cleanliness standards were maintained to ensure a high-quality assembly of the reactor and main circuits.

After assembly the sodium loops were not flushed with water. The main circuits were dried and degassed by evacuating and simultaneously heating the equipment and piping to 200-250°C. During the drying process the sodium circuits were periodically filled with pure argon. The ultimate vacuum attained in the primary circuit was $2 \cdot 10^{-1}$ mm Hg with an inleakage of $0.8 \cdot 10^{-1}$ mm Hg per hour.

A procedure different from that used on the BR-5 was developed at the BOR to prepare sodium for the coolant circuits without distilling it. Each transport container of sodium (volume 1 m^3) was heated to 200-250°C for 3 h to remove the paraffin by vacuum distillation. After this the sodium was transferred from the transport container under argon pressure through a mechanical mesh filter into a 6 m^3 intermediate tank where it remained for some time, and then into a 35 m^3 receiving tank. The final purification of the sodium was performed by circulating it through a cold trap. The main impurities in the original sodium are: carbon $4.5 \cdot 10^{-3}$%, hydrogen $6 \cdot 10^{-4}$%, nitrogen $1.5 \cdot 10^{-3}$%, calcium $6 \cdot 10^{-3}$%, and potassium $2 \cdot 10^{-2}$%. The oxide content of the sodium corresponded to a plugging temperature of 120°C.

After the primary circuit was filled with sodium it was heated to 250°C. During the heating the strains in the reactor vessel, the equipment, and the piping were checked with strain gauges. The strains did not exceed admissible values. Because of the care given to maintaining clearliness during assembly the oxide content of the sodium rose only to t_{plug} = 180°C when the primary circuit was filled. Other contaminants were present in practically their original proportions. Contaminants were not detected on the mesh filters placed in the piping, or in the fuel assembly mock-ups. After filling the primary circuit with sodium, loading the core, and replacing the accessories, the total amount of sodium oxides transferred into the trap was only 2 kg.

The physical and power start-ups of the BOR reactor took place at the end of 1969 simultaneously with the removal of heat by an air heat exchanger. During 1970 extensive research was performed on the physical, thermal, and hydraulic characteristics of the equipment. Measurements of the physical characteristics of the reactor showed that all basic physical parameters had been predicted accurately enough [4]. This applies to such quantities as the critical mass, the effectiveness of controls, and the effect of temperature on reactivity. The experimental coefficients of nonuniformity of heat release and the decrease in reactivity with burnup turned out to be somewhat smaller than the calculated values. On the whole the good predictions of physical characteristics of the BOR reactor were due to greater experience in making calculations and to the studies of a model of the reactor on the BFS test rig [5, 6]. The experimental program on the BFS rig confirmed the method and constants for the physical design and permitted refinements in the predicted parameters, particularly in the effectiveness of controls.

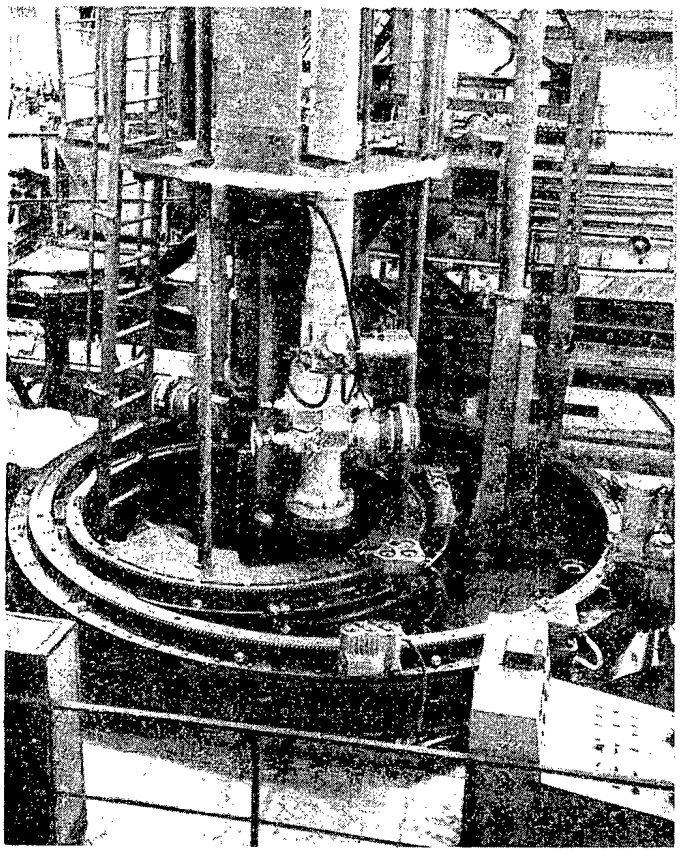

Fig. 1. BN-350 reactor test rig. Large and small rotating plugs.

In the BOR reactor, as in BR-5, in the first stage of operation at reduced sodium temperatures and negligible powers, up to 2-3 liters of argon accumulated on the core surface. This manifested itself in a change in reactivity from the increased pressure in the gas space and in an increase in coolant flow rate. It was observed later that this effect disappears after the reactor has been operated at high power but shows up again to a negligible degree after long shutdowns. An increase in coolant flow rate from 0 to 800 m^3/h leads to a positive reactivity of ~0.2%. An analysis of the radioactive contaminants in the primary circuit indicates the integrity of the fuel element cladding. When the reactor operates at 40 MW the sodium activity is determined by Na^{24} (38 Ci/kg), and the after-shutdown activity is due to Na^{22} ($4.7 \cdot 10^{-5}$ Ci/kg); the other activated isotopes are mainly contaminants and their total activity is less than $3 \cdot 10^{-5}$ Ci/kg. In the gas space the activity of Ar^{41} is $3 \cdot 10^{-3}$ Ci/liter, and Ne^{23} is 1 Ci/liter; radioactive isotopes of xenon and krypton (~10^{-6} Ci/liter) result from surface contamination of the fuel elements.

Studies of the thermal hydraulic characteristics of the reactor also showed good agreement between experimental and design parameters. Measurements of sodium flow rates through core and radial blanket assemblies were made with special apparatus. Differences in sodium flow rates through assemblies in the same orificing zone did not exceed 5%.

The fraction of the sodium cooling the blanket assemblies, the USZ, and the reactor vessel was larger than normal initially and was decreased somewhat by special control elements. The temperature of the sodium leaving the core and blanket assemblies was measured with 38 thermocouples placed over the assemblies. An appreciable fraction of the thermocouples showed a temperature drop, apparently as a result of cross flow of cold sodium over the ends of the assemblies.

In a test of the circulation of sodium through the primary circuit, a vibration of the check valves was observed when the pumps were not started or stopped simultaneously. Study of the loop vibrations showed that the frequency was related to the angular speed of the pump; the amplitude of the vibrations was small, but there were resonance frequencies. Measurements permitted the selection of operating ranges of pump speeds for minimum vibration. The basic technological reactor equipment – pumps, heat exchangers, oxide traps, etc. – operated well.

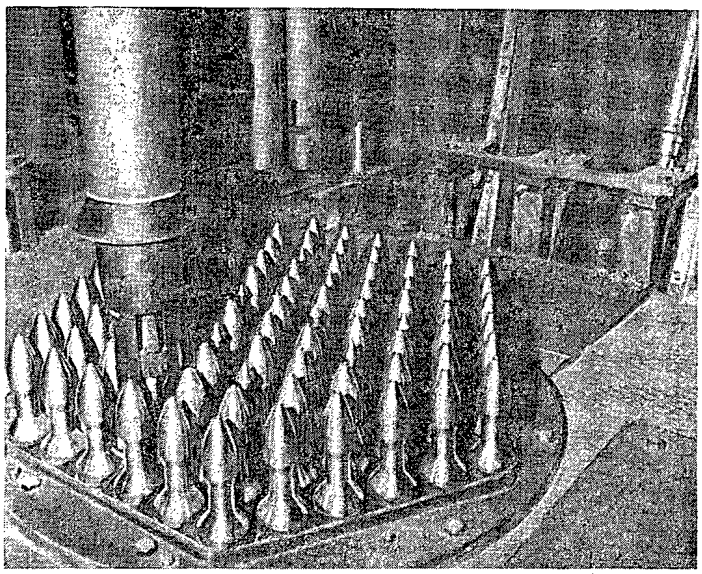

Fig. 2. BN-350 reactor test rig. Core mock-up with refueling mechanism.

Late in 1970 a once-through coil type steam generator with single-wall separation of sodium and water was started up. Steam from the steam generator at a pressure of 85 atm and a temperature of 430°C was fed to the turbine which was connected to the power system. At the present time the BOR reactor is operating at 40 MW with a mean sodium temperature of 500°C at the apparatus outlet.

BN-350 Reactor

In the town of Shevchenko assembly tasks are being completed and preparations are being made for the start-up of an atomic power plant with the BN-350 reactor. In the early part of 1971 the reactor vessel, the pump housings, the heat exchangers, and the steam generators were assembled, and the most complex and large scale elements were conducted and put in place: the rotating plugs, the pressure header, and the elements of the fuel loading and unloading system, etc.

Complex experimental and theoretical studies were made to confirm the design. We present below the results of experimental studies of individual subassemblies and complex mechanisms which supplement results published earlier [7-9].

In order to produce reliable components of the refueling devices tests were made of various materials to determine their mechanical properties under operating conditions.

Special attention was paid to the selection and test of materials to be used at point of sliding or stationary contact. The behavior of materials was studied over a wide range of contact loadings and for long times. Various combinations of materials were investigated, in particular 0Kh18N10T and Kh18N9 steels with and without cladding of contact surfaces, operating in sodium at temperatures up to 550°C. As a result of these tests structural changes were made in certain units of the mechanisms operating in the high-temperature regions, or the operating conditions were changed.

"Hot" tests of the refueling mechanism and the elevator were performed on test rigs with circulating sodium. The basic parameters of the rigs were: sodium temperature 500°C with mechanisms turned off, and 300°C with mechanisms operating; oxygen content of sodium up to $50 \cdot 10^{-3}$ wt. %; volume of liquid metal in each rig 1 m^3. The "hot" tests included the study of artificially created "accidents": failure of the control system, complete loss of power, increase in the oxygen content of the sodium, etc. During the tests the refueling mechanism performed more than 5000 operations of removing and inserting fuel assembly mock-ups, and the elevator completed about 3000 round trips. Tests of the pilot models of the refueling mechanism and the elevator proved the capabilities of these mechanisms under normal operating conditions and their reliability in emergencies.

A test rig (Figs. 1, 2) was constructed to test the combined operation of the refueling mechanism and its controls, the system guiding the refueling mechanism onto the head of a fuel assembly, and the locking

device. The test rig included the following: metal structures simulating the vessel and the reactor transfer box, the rotating plugs, the central column with the SUZ tubes, the elevators, the refueling and fuel element transfer mechanisms, mock-ups of the fuel assemblies and the pressure header, electrical equipment and guiding systems. In tests of the complex refueling mechanisms a number of refuelings with fuel assembly mock-ups were carried out by using the regular mechanisms and control systems. The system was tested under normal conditions and in the following situations simulating accidents: 1) inserting and removing mock-ups of fuel assemblies bowed up to 10 mm and rotated by 60, 120, and 180°; 2) interchange of assemblies with the axis of the refueling mechanism displaced 10 mm from the axis of the assembly head; 3) test of the possibility of release and withdrawal of the refueling mechanism from a stuck assembly raised 500 mm above the core. The experiments proved the capabilities of the mechanisms under the conditions indicated.

In order to guarantee a good fit of the vessel when put together at the construction site, a complex test assembly of the vessel was made on a special rig at the manufacturer's works; the whole vessel was held together by special fixtures and temporary welds of the circumferential seams. The assembly confirmed the accuracy of manufacture of the units and showed the vessel to be leak-tight. The units of the vessel were delivered to the assembly site and the final assembly of the vessel was completed between August and October of 1968. After the vessel was assembled, welded, and carefully inspected it was subjected to hydraulic and vacuum tests. Strain gauges enabled the state of stress of the vessel to be studied during testing.

To confirm the accuracy of manufacture of the neutron shield support and pressure header and also their fit with the core, a test assembly of these elements was made at the manufacturer's works and the necessary tests performed. The assembly used the regular pressure header, parts of the neutron shield support, and the control rod guide tubes, but mock-ups of the fuel assemblies. The mock-ups matched the actual assemblies exactly in external dimensions, weight, and construction of the shaft and head. In the test assembly each fuel assembly mock-up was set in a socket of the pressure header under its own weight.

Most of the experimental reactor physics studies of the BN-350 were made on the BFS rig. Several critical assemblies differing in the positions of the controls were used. Some of the assemblies had a homogeneous core without controls. The basic experimental data obtained with these assemblies are given in [10-12]. A more careful analysis of the experiments and the revision of the constants for the principal isotopes U^{235} and U^{238} showed that the critical parameters of the reactor were corrected with the previously published data. As a result a new critical size of the BN-350 was determined. The new nuclear data led to a better agreement between the calculated and measured effectiveness of boron rods than had been achieved earlier by using the 1964 26-group constants. The experiments showed that the heat release distribution over the reactor core had been calculated accurately enough.

Experimental studies of the Doppler effect were performed on the BR-1 reactor by the method of activation of heated and cold foils in a uranium oxide prism. The experiments showed that the temperature dependence of the U^{238} capture cross section had been taken into account accurately enough in calculating the Doppler effect. However, the Doppler coefficient of reactivity in a reactor, taking account of all uncertainties, can hardly be calculated to an accuracy of more than 20-30%. Measurements of the central reactivity coefficients, the spectral characteristics, and the prompt neutron lifetimes in various modifications of the BN-350 assembly indicate that predictions of the details of the neutron spectrum and the neutron importance cannot be made accurately enough, and show the necessity of improving both the constants and the calculational methods.

BN-600 Reactor

The structural and design tasks on the BN-600 have been completed. At the present time the construction of the buildings is well along, and the manufacture of equipment at the factories has begun. Following the usual procedure experimental studies to substantiate and confirm the characteristics of the equipment are being made.

The BN-600 reactor is the next stage after the BN-350 on the way to producing high-power and economically efficient atomic power plants with fast reactors. Technical and economic analyses show that to produce a fast-reactor power plant with a high economic index it is necessary to force the steam parameters in comparison with the BN-350 parameters. From these same considerations it is expedient to increase the power per reactor and the degree of fuel burnup. Operating requirements of the station in the

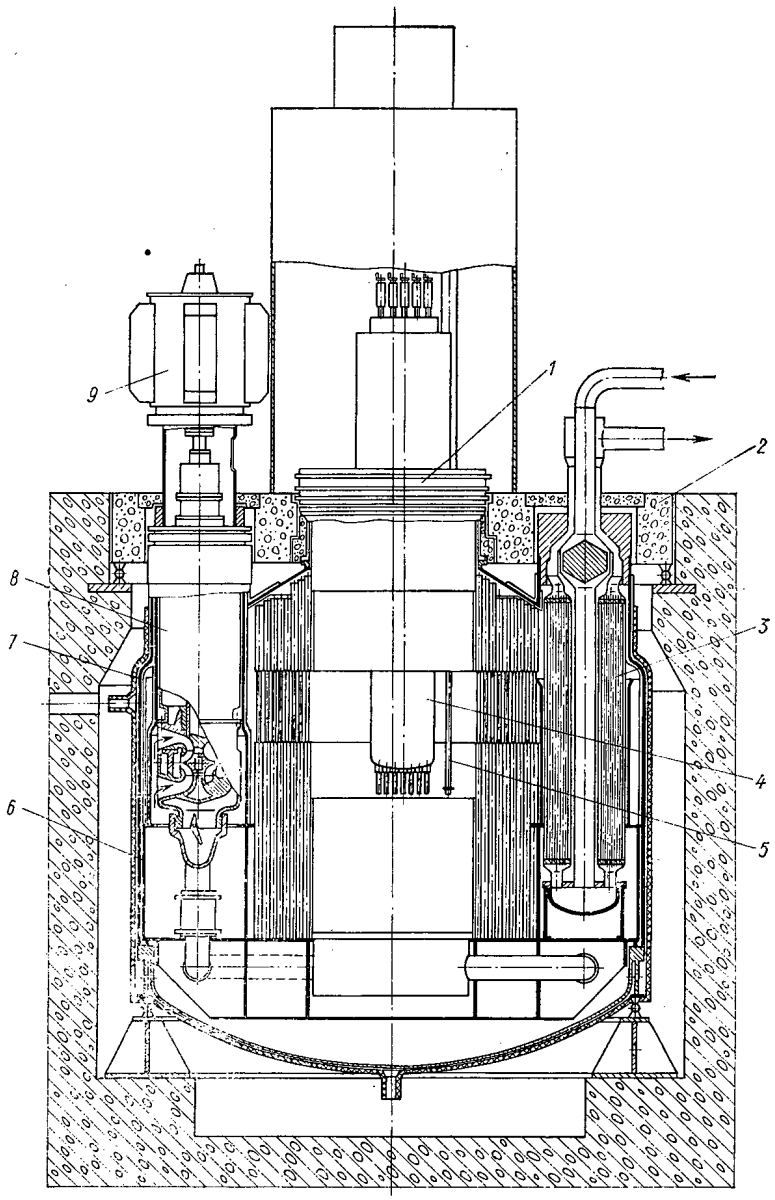

Fig. 3. Longitudinal section through BN-600 reactor. 1) Rotating plugs; 2) upper stationary shield; 3) heat exchanger; 4) central column with SUZ mechanism; 5) refueling mechanism; 6) supporting frame; 7) vessel; 8) pump; 9) electric motor.

power system require increasing the time between refuelings. These factors form the basis for the development of the BN-600.

The choice of temperature and hydraulic characteristics of the liquid-metal circuits and the thermodynamic parameters of the steam–water cycle of the BN-600 reactor was based on the following considerations. The use of supercritical parameters presents great difficulties in developing steam generators; the capital expenditures for the high-temperature steam-power part are increased. Therefore parameters which ensure an efficiency of 42-43% for the steam turbine cycle with a fuel element cladding temperature of 680-700°C turned out to be more acceptable. Analyses showed that such compromise parameters are: sodium temperature at reactor outlet 550°C, mean temperature rise of sodium in reactor 170°C, steam parameters at admission to the turbine 500°C and 130 abs. atm.

It should be noted that the use of the well-developed K-200-130 series of turbines ensures an efficiency of up to 43% for a turbine plant with the indicated steam parameters.

The data accumulated so far permit a core design with the temperatures indicated above and fuel elements containing a mixture of uranium dioxide and plutonium dioxide in stainless steel cans with a 10% burnup of the heavy atoms. However, the problem of the swelling of steel at such a burnup has not yet been adequately studied. Sealed fuel elements are used in the BN-600. In order to lower the gas pressure a gas space 0.7 m long is provided in the lower part of an element for a fueled length of 0.75 m. A number of factors were taken into account in choosing the core size, shape, and composition. Studies showed that in practice it is very difficult to increase the volume fraction of the fuel to 45-47% and higher. In this respect the composition of the BN-600 core is not very different from that of the BN-350. The BN-600 core is rather highly compressed (D/H = 2.7). Such a core configuration makes it possible to increase the coolant flow rate through the reactor with a moderate hydraulic resistance and to decrease the heating. The difference between the maximum cladding temperature and the temperature at the reactor outlet is decreased.

It is important to be able to operate a power reactor for a long time between refuelings, and it is desirable to have refueling operations coincide with planned shutdowns of the atomic power plant. An effective system of reactivity compensation enables the BN-600 reactor to operate for four to five months at full power between refuelings. The use of a large number of absorbing rods led to a search for their optimum arrangement, since the heat release distribution is very sensitive to an asymmetry of the rod arrangement. The BN-600 reactor uses a dispersed system of rods which minimizes the variations in heat release between the outer position of the compensators and also permits the use of natural boron and tantalum as absorbing materials.

The BN-600 reactor was designed to operate in a tank. This construction was chosen mainly for the following two reasons: 1) the compact arrangement of the primary circuit equipment and all the radioactive coolant in one tank eliminates the necessity for separate rooms and simplifies the problems of maintaining a leak-tight primary circuit; 2) in order to make a sound choice of the optimum structural design for high-power reactors it is necessary to accumulate operating experience with both loop (BN-350) and tank (BN-600) type reactors.

In the course of designing the BN-600 reactor several alternate arrangements of the primary circuit components were developed; everything in one tank; core and heat exchangers in the tank with the pumps outside the tank and connected to it by special piping; pumps and heat exchangers in a separate vessel, etc. Figure 3 shows the version adopted for the reactor. A detailed description of it is given in [13]. Peculiarities of the arrangement of the equipment in the tank raise a number of special problems: 1) the transfer of loads and the mounting of the reactor tank, core, heat exchangers, pumps, rotating plugs, and other internal reactor elements; 2) vibrations of equipment and structural elements; 3) compensation of thermal expansions of interconnected components inside the tank; 4) operation of reactor tank to ensure elimination of thermal shocks and nonuniform temperature distributions around the perimeter and over the height; 5) hydrodynamics problems in the tank: uniform distribution of coolant flow through the heat exchangers; elimination of entrapment and pumping over of gas through the circuit etc.

The following schemes for transferring loads and supporting components were analyzed:

1) The whole internal reactor structure – core and shield, pumps, heat exchangers, rotating plug – is mounted on a lower supporting frame (Fig. 3). The load on the frame is transferred to the foundation through special supports. The lower frame with the equipment mounted on it and the upper part of the reactor are bathed in coolant at different temperatures. Thus one of the difficult problems in this construction is to maintain equality of the thermal expansion of the upper part of the reactor and the equipment mounted on the lower frame, and to ensure leak-tight joints between the tank and the external piping. These problems were solved in the BN-600 reactor by using bellows at the pump and heat exchanger connections. Cooling of the tank, even mainly the upper part of it, decreases the difference in the transverse displacements of the upper reactor cover and the supporting frame on which the heat exchangers and pumps are mounted, and thus improves operating conditions of the bellows which are located in the gas space so they can be replaced. The rotating plugs together with the central column and the refueling mechanism are attached to the upper tapered cover of the tank. The thermal expansion takes place upward from the roller bearings on which the reactor is mounted. The load on the rollers is transferred to the foundation.

2. The reactor tank and a special frame supporting the core are hung from a heavy load-bearing plate. The thermomechanical equipment is placed on an upper plate. To keep the equipment on the plate stationary in this arrangement requires bellows 2.0-2.5 m in diameter operating in the coolant, or movable

supports in combination with bellows. The upper plate turns out to be heavily loaded and complicated in structure. Therefore the first version is preferable.

A number of structural units used and experimentally validated in building the BN-350 were utilized in the construction of the BN-600: the rotating plugs and their tin–bismuth alloy seals, the refueling mechanism, the elevator, the basic elements of the fuel assembly transfer mechanism etc. Retention of the external dimensions of the fuel assemblies permits the use of the existing technology of tube construction and the takeover of units of the transfer-process part of the BN-350 equipment. All this appreciably decreased the amount of experimental testing. The planned program of experimental work provides for tests of full-scale pumps for the primary and secondary circuits, a model of the heat exchanger, a module of the steam generator, hydraulic reactor tests on a 1:6.6 scale model etc. Some of the planned experimental work has already been completed.

Strength tests were performed on models of the most heavily loaded components of the vessel: the cover and the supporting frame. The models were made of sheet steel on a 1:10 scale and constructed to simulate the rigidity of the units. The tests showed that the vessel cover and the supporting frame were strong enough for the maximum loads including the static loads which were twice nominal. Various types and constructions of steam generators were developed in the design process. A once-through steam generator of the modular type was chosen for the BN-600. Type 1Kh2M low-alloy perlitic steel is used as the structural material in the evaporator, and type 18/8 austenitic steel in the superheaters.

The BFS-2 test rig was constructed at the Physics and Power Institute to study the physical characteristics of the BN-600 reactor and higher power fast power reactors. The dimensions of the rig permit a full-scale mock-up of large reactors. At the present time the BFS-2 rig is being used to study a model of the BN-600. The main purpose of the studies is to obtain experimental data on the critical characteristics, the effectiveness of the burnup compensation and safety systems, and the heat release distribution in the presence of the compensating rods. In 1970 experiments were performed on the BFS-2 rig to study the penetration of neutrons through a mock-up of the reactor shield inside the vessel. The BFS-2 is air cooled; experiments with a plutonium loading can be performed on it.

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DESIGN CHARACTERISTICS AND OPTIMIZATION OF THE REACTIVITY MODULATOR IN THE IBR-2 REACTOR

V. D. Anan'ev, D. I. Blokhintsev,
V. V. Bondarenko, Yu. M. Bulkin,
E. N. Volkov, G. N. Zhemchuzhnikov,
V. P. Zinov'ev, B. I. Kuprin,
V. S. Lobanov, A. K. Popov,
V. S. Smirnov, V. I. Sokolov,
F. I. Ukraintsev, I. M. Frank,
A. I. Khoperskii, N. A. Khryastov,
E. P. Shabalin, and Yu. S. Yazvitskii

A pulse fast reactor (PFR) was put into operation in 1960 at the Joint Institute of Nuclear Research [1]. This is a new type of reactor which generates periodic power pulses lasting tens of microseconds by mechanical modulation of reactivity. Immediately after it was put into operation, the reactor was used for physical experiments, in which the basic characteristics of such a reactor as a neutron source were determined. Since it generates neutrons in periodic pulses, the PFR can be used for experiments based on flight-time methods. During the past decade, the reactor was used mainly for neutron spectrometry in the energy range from millielectron volts to kiloelectron volts, i.e., in the energy range which is of special interest for nuclear physics and the physics of condensed media. In these investigations, while it operated in its initial variant at a time-averaged power level of a few kiloelectron volts, the PFR complemented, and sometimes proved to be more efficient than, the pulse neutron sources based on accelerators and stationary reactors with mechanical choppers which were then in use. The experience gained in PFR operation has shown that, for experiments requiring pulsed neutron fluxes, periodic pulse reactors provide a high recording intensity at much lower power levels in comparison with stationary reactors. Thus, in resonance neutron measurements, the PFR at 3 kW was equivalent to a stationary reactor with a power level of 20-30 MW [2]. For experimental purposes, the saving of power results in a very low background, which means that exceedingly weak effects can be detected. From the technical point of view, a low mean power level simplifies the design of the reactor and its technological systems, ensures a compact core, and reduces the operating costs.

The advantages of periodic pulse reactors are also obvious in the production of large neutron fluxes. The largest modern experimental reactors (for instance, SM-2 in the USSR and HFBP in the USA) produce fluxes of $\sim(2\text{-}3)\cdot10^{15}$ neutrons/cm$^2\cdot$sec. Such fluxes probably constitute the technological limit for stationary reactors. In any case, in order to increase the flux to 10^{17}-10^{18} neutrons/cm$^2\cdot$sec, a stationary reactor with a power level of thousands of megawatts or more would be necessary, which would be inadvisable because of the high cost of equipment, operational difficulties, danger of contamination by reactor waste, etc. Periodic pulse reactors produce fluxes of the order of 10^{18} neutrons/cm$^2\cdot$sec per pulse at a mean power level of a few megawatts (for instance, IBR-2). In principle, they are capable of producing high-intensity fluxes at reasonable mean power levels. Considering single-pulse reactors with regard to the magnitude of pulsed fluxes (such reactors produce fluxes up to $5\cdot10^{19}$ neutrons/cm$^2\cdot$sec), we find that they are inferior to periodic pulse reactors with respect to the integral neutron dose. Thus, IBR-2 produces approximately $3\cdot10^{19}$ neutrons/cm^2 per day in the internal channel, while a single-pulse reactor produces only about 10^{16} neutrons/cm^2.

State Committee for Utilization of Atomic Energy of the USSR. Translated from Atomnaya Énergiya Vol. 31, No. 4, pp. 352-358, October, 1971.

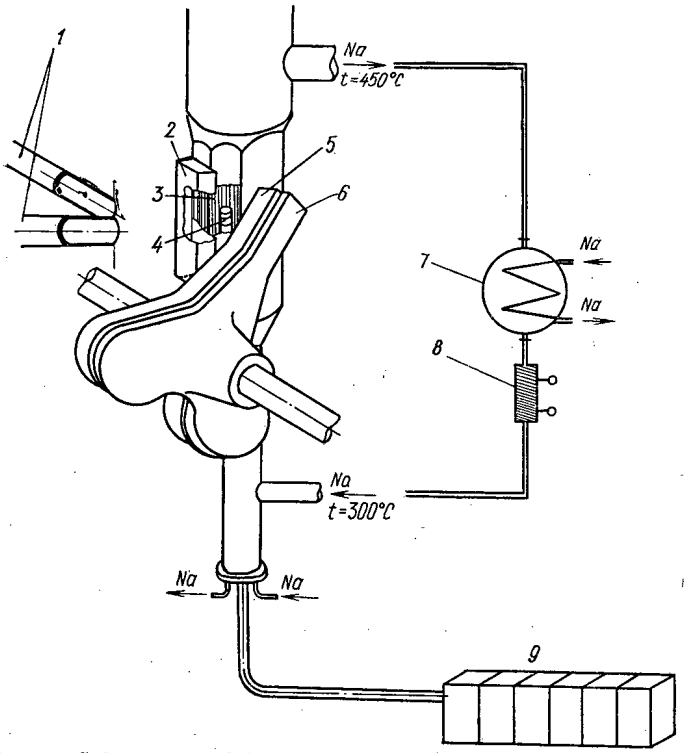

Fig. 1. Schematic of IBR-2. 1) Experimental channels; 2) aqueous moderator; 3) core; 4) tungsten target; 5) mobile reflector; 6) auxiliary mobile reflector (AMR); 7) heat exchanger; 8) electromagnetic pump; 9) linear accelerator.

These considerations indicate the usefulness of pulse reactors for investigating the dynamics of radiation damage, short-lived isotopes and isomers, etc.

The booster operating conditions of the PFR, when the reactor in the subcritical state acts as a breeder of neutrons from an external pulse source, which are injected at the time of maximum reactivity, are of great interest for neutron spectrometry at energies exceeding a few electron volts (investigations in nuclear physics and physics of condensed media, especially when the latter are acted upon by strong pulsed magnetic fields or pulsed high pressure). In contrast to stationary boosters (the type used in Harwell), a pulse booster, which operates mainly on prompt fission neutrons, ensures a high multiplication constant (a few hundreds or more) with a lower background between pulses. Pulse booster conditions in a PFR were first realized in 1964 [3]. This mode of operation improved by a factor of more than 100 the quality of the source, which is characterized by the ratio of the mean yield to the square of the pulse duration (parameter Q/τ^2). A pulse reactor operating under booster conditions is suited for operation with pulse durations of a few microseconds. In nuclear physics, such devices are used for investigating low-probability processes, the composition of neutron-induced radiation, various correlations, etc. In such work, a neutron pulse duration of a few microseconds constitutes a reasonable compromise between the requirements for pulse intensity and the resolving power of the device.

The start-up of the first PFR in 1960 proved the feasibility of controlling a chain fission reaction on prompt neutrons, which had been considered doubtful. The design of the PFR proved to be convenient and reliable. Without substantial modifications, the PFR operated until 1968 at a mean power level of 1-6 kW. Besides the above-mentioned operation, a new set of operating conditions with "rare pulses" was introduced in 1968 [3], where pulses with a duration of 36 μsec were generated at a frequency of 0.2 pulse/sec. The pulse power reached 10^6 kW.

During the operation of the PFR, it was found possible to raise the power level (without modifying the general PFR layout or enlarging the core) by improving the air-cooling system and providing two active inserts for reactivity modulation (instead of a single one in PFR's). Improvement of the PFR design along these lines resulted in the construction of a new reactor in 1969, IBR-30. This reactor is combined with

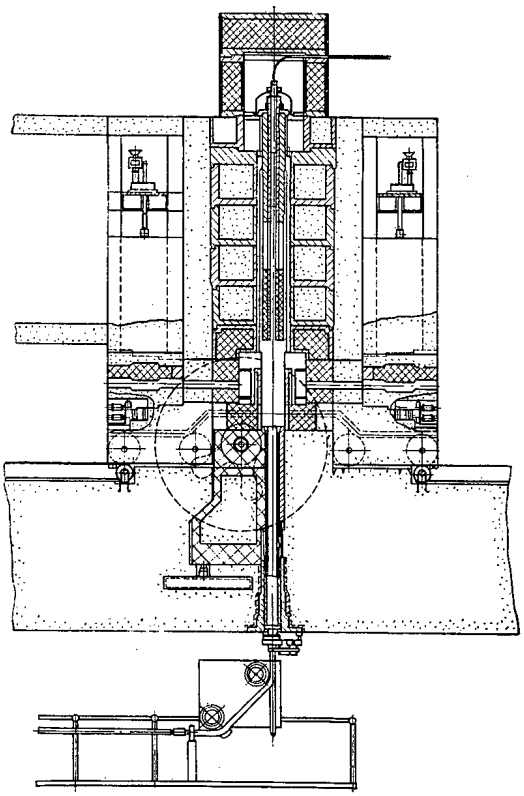

Fig. 2. Vertical section of IBR-2.

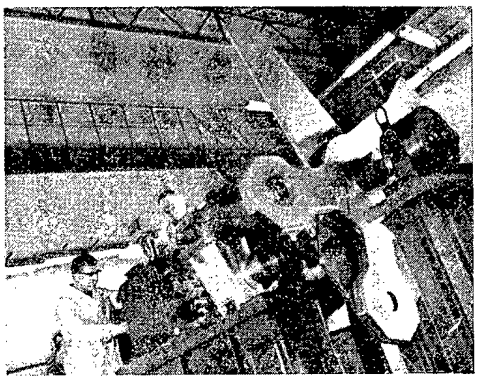

Fig. 3. Experimental reactivity modulator (without the sealing frame).

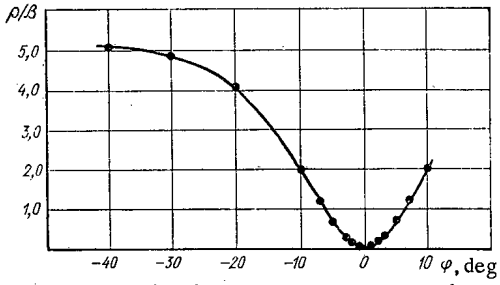

Fig. 4. Multiplication constant as a function of the MRE shift.

the LIU-40 linear electron accelerator. The power level of IBR-30 presently reaches 25 kW and can be raised even more. Under booster conditions, IBR-30 operates at a level of approximately 2.5 kW, producing pulses with an approximate duration of 3 μsec. The rising interest in neutron-spectrometry in many branches of science and the exacting demands for intensity have stimulated the development of the new IBR-2 megawatt reactor with an LIU-30 high-current injector.

The IBR-2 reactor has a compact core with concentrated nuclear fuel (plutonium dioxide) similar to the BR-5 core, which has been tested in operation. The high specific power is achieved as a result of liquid-metal sodium cooling. Pulsed operation of the reactor is achieved by changing periodically the reactivity of the system and moving a part of the neutron reflector relative to the core (Fig. 1).

The design of the reactor and its equipment for physical experiments has been described in detail earlier [3, 4]. The basic characteristics of the IBR-2 reactor are the following:

Reactor (mean) power level	4 MW
Pulse power	
at 5 Hz	7700 MW
at 50 Hz	700 MW
Core volume	20 liters
Fuel, PuO_2	88 kg
Lifetime of fission neutrons	$4.2 \cdot 10^{-8}$ sec
Pulse duration	90-100 μsec
Fast neutron flux at the core center (at the maximum of a pulse)	$1.1 \cdot 10^{18}$ neutrons/cm^2 · sec
Thermal neutron flux in the moderator (at the maximum of a pulse)	10^{17} neutrons/cm^2 · sec
Time-averaged thermal neutron flux from the moderator surface	$1 \cdot 10^{13}$ neutrons/cm^2 · sec
Power level under booster conditions at 50 Hz	$0.14 \cdot \theta$ MW (θ is the pulse duration, μsec).

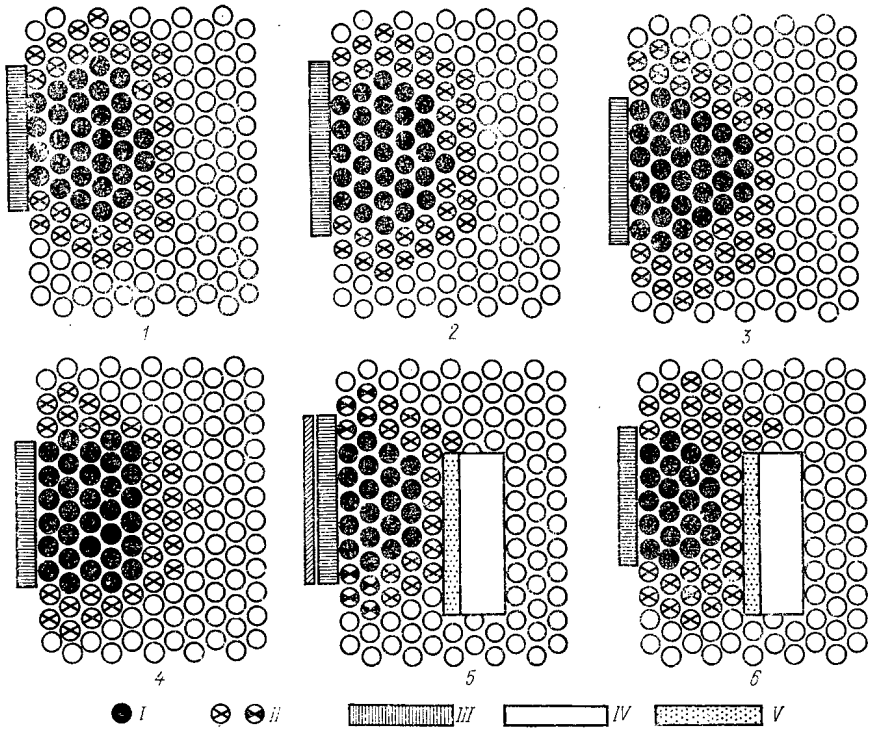

Fig. 5. Recorder chart of the loading of IBR-2 simulators. 1-6) Variants of the core geometry: I) core rods; II) rods of the stationary reflector; III) MRE; IV) AMR; V) hydrogenous moderator.

The reactivity modulator (RM) of IBR-2 (Figs. 2 and 3) constitutes a basically new element in reactor construction. A two-rotor modulator is used for regulating the pulse frequency while maintaining a constant pulse width. The rotor of the basic reflector has three blades, one of which (radius, 1200 mm) is the mobile reflector (MRE) proper and has a 24 × 7 cm cross section, while the other two are counterweights. The rotor is made of high-strength steel. Tungsten weights are provided in the counterweight in order to reduce their size and provide the possibility of adjusting the position of the center of gravity. The rotors of the basic and the auxiliary reflectors are mounted coaxially and are driven by the same motor. The rotor of the basic reflector turns at a constant speed (3000 rpm), while the auxiliary reflector (AMR) is driven by means of a gear box. This facilitates control of the repetition frequency of power pulses at 50, 25, 10, and 5 Hz, since a pulse is produced only at the time of simultaneous passage of the basic and the auxiliary reflectors near the core. Stable, controllable slippage of the driving shaft relative to the driven shaft is provided. A 10% slippage corresponds to frequencies of 45, 22.5, 9, and 4.5 Hz. The rotation of a heavy, complex steel rotor near the core imposes stringent requirements for the structural reliability of the modulator. The situation is aggravated by the fact that the MRE can cause a sudden change in the insertion reactivity and, consequently, in the pulse energy. Therefore, besides checking the design by calculation, the stressed state of the basic reflector was simulated on parts made of optically active resins.

Figure 3 shows the first variant of the RM. The program of RM investigations calls for determining the actual stressed state of rotors of the basic and the auxiliary reflectors, vibrometer measurements of the supports and blades of the MRE, and continuous monitoring of displacement of the MRE rotor. A sudden axial shift of the rotor can occur as a result of failure of the roller bearings in the rotor supports. This shift would consist of the axial play of the roller bearings, the radial clearance between the shaft of the reflector rotor and the auxiliary bearing.

The design of the RM in IBR-2 allows a shift of 0.9 mm. Under certain conditions, such a shift of the MRE can produce a reactivity jump $\Delta\varepsilon = 0.00053$, which would correspond to a pulse energy of $1.5 \cdot 10^4$ kJ. Therefore, in mounting the RM supports, the axial play of the rotor bearings must not exceed 0.1 mm. In order to ensure reliable operation of the bearings, they will be placed behind a radiation shield consisting of a tungsten and tungsten boride layers. The shield is designed for operation over a period of two and a half

TABLE 1. Efficiency of the MRE for Different Cores

Variant of the core geometry	MRE dimensions, cm	Δk, %	α, $\times 10^{-4}$ cm^{-2}
1	35×7×45	3,8±0,5	0,76±0,01
	30×7×45	3,63±0,12	0,88±0,03
	25×7×45	3,24±0,09	1,06±0,01
	20×7×45	2,88±0,10	1,03±0,08
2	35×7×45	7,7±0,5	0,99±0,03
	30×7×45	7,5±0,3	1,30±0,08
	25×7×45	5,66±0,33	1,20±0,07
	20×7×45	5,54±0,24	1,04±0,05
3	35×7×45	4,98±0,03	1,13±0,05
	30×7×45	4,98±0,01	1,17±0,03
4	35×7×45	5,6±0,1	1,16±0,07
	30×7×45	5,2±0,2	1,19±0,02
	25×7×45	4,7±0,2	1,10±0,02

TABLE 2. Effect of Thickness of the MRE on Its Efficiency

MRE thickness, mm	α, $\times 10^4$ cm^{-2}	Δk, %
40	0,72±0,08	3,03±0,13
70	1,2±0,1	4,9±0,2
100	1,56±0,06	5,02±0,24

TABLE 3. Effect of Width of the MRE on Its Efficiency

MRE thickness, cm	α, $\times 10^{-4}$ cm^{-2}		Δk, %	
	inclined 26°	vertical	inclined 26°	vertical
20	1,23± ±0,04	—	4,00± ±0,12	—
25	1,26± ±0,11	1,38±0,04	4,87± ±0,19	4,80±0,22
30	1,07± ±0,05	1,38±0,02	5,44± ±0,22	5,10±0,14

TABLE 4. Comparison between Theoretical Data and Experimental Results

Assembly variant	MRE width, cm	α, $\times 10^{-4}$ cm^{-2}	
		experiment	calculation
1	30	0,88±0,03	0,90±0,18
4	35	1,16±0,07	1,07±0,20
5	35	1,56±0,07	1,50±0,30
6	25	1,38±0,04	1,70±0,30

years with an integral prompt neutron flux of $2.5 \cdot 10^{20}$ neutrons/cm^2 ($E \approx 1$ MeV). The shield ensures a 1300 h lifetime of the lubricating oil in a forced-circulation system. Because of the complex operating conditions and the peculiarities of the radiation conditions, the RM has been constructed in units that can be disassembled by remote control.

The critical assembly of the simulator IBR-2 was completed in 1968 on the large physical stand (LPS) at the Physics and Power Institute [5]. The present paper discusses the assembly results only from the point of view of the optimum geometry of the core and the MRE.

The specific design features of the LPS prevented exact simulation of the IBR-2 composition. Highly enriched U^{235} was used as fuel instead of plutonium, and U^{238} was used for the stationary reflector instead of tungsten. Considering that the reactivity effects investigated on the the assembly depend mainly on the geometry, this simulator is adequate for the problem in question.

The core of the simulated reactor occupied a small part of the LPS tank; it was composed of pipes with a diameter of 50 mm, filled with the necessary materials in the shape of tablets. The fixed reflector was constructed in the same manner. A special mechanical system was used for simulating the rotating (basic and auxiliary) reflectors. The lever of the reflector could rotate slowly about the horizontal axis from −10 to +40°. The remote control inspection system made it possible to place the mobile reflector unit at any point within this sector with an accuracy to ±2 mm (the path length is measured at the mean radius of the MRE unit). The drive of the reflector lever could be mounted on the support beam in two positions: vertical mounting on the mobile reflector and inclined mounting at 26° with respect to the vertical axis of the core. The latter installation of the reflector disk was used in the IBR-2 design. A steel plate with a thickness of 7 mm, which simulated the reactor vessel, was placed between the core and the mobile reflector.

The typical behavior of k_{eff} with the MRE shifting along an arc is shown in Fig. 4. For almost all the MRE's investigated, the curve is parabolic within ±5 cm from the symmetric position:

$$k(x) = k(0) - \alpha x^2.$$

Four core variants were considered in investigating the effect of the cross-sectional shape of the core on the MRE characteristics (Fig. 5). Table 1 provides the measurement results for the parabolic reactivity coefficient of the MRE (α) and the effect of moving away the MRE by 40° (Δk) for four assembly variants and different MRE dimensions. It is evident that the maximum values of α and Δk are obtained with the second core variant when the dimension of the core side nearest to the MRE (the dimension of the core "window") is close to the MRE width.

The coefficient α decreases very slowly with an increase in the MRE width, while the total efficiency of the MRE (the value of Δk) increases still further. Therefore, the optimum width of the MRE is 1.1-1.2 times as large as the "window" width. A special, the fifth, assembly variant with a 14 liter core was designed for investigating the characteristics of the IBR-2 simulator with concentrated nuclear fuel. This

assembly was used for investigating the effect of the thickness of the MRE on its efficiency. The results are given in Table 2. It is evident from these data that α and Δk do not change appreciably if the MRE thickness is increased by more than 10 cm.

The effect of axial shifts of the MRE on the reactivity of the system was investigated. It was found that the dependence of α and Δk on the clearance between the MRE and the wall simulating the reactor vessel is linear in the 2-22 mm range; the relative change amounts to 30% when the MRE is shifted by 20 mm. The reactivity effects connected with the axial shift of the MRE are important for estimating the pulse power fluctuations caused by vibrations of the mobile parts. Using experimental data, we can estimate the fluctuation of the pulse energy; it amounts to 20-30% when the amplitude of axial oscillations of the MRE is equal to 0.1 mm. The efficiency of a $45 \times 35 \times 3.4$ cm auxiliary mobile reflector (AMR), mounted behind the MRE ($45 \times 35 \times 7$ cm), was $(0.68 \pm 0.04)\%$.

The sixth assembly was used for investigating the characteristics of the IBR-2 simulator with diluted fuel (core volume, 20 liters). The cross section of the core was identical to that of the fifth assembly, while the core height was 45 cm instead of 32 cm.

This assembly was used for measuring the MRE characteristics with the lever in the vertical position and at an angle of 26° to the vertical. The values of α and Δk (the thickness and height of the MRE are equal to 7 and 45 cm, respectively) in dependence on the width and angle of the MRE are given in Table 3.

The total effect of moving away the AMR with the vertical lever was $(0.91 \pm 0.03)\%$. The reactivity change due to the axial shift of the AMR amounted to $0.14 \cdot 10^{-2}$ cm^{-1}.

The physical characteristics of the IBR-2 assembly were calculated by using the Monte Carlo method for the three-dimensional geometry. Good agreement with experimental data was observed for most parameters [6]. The theoretical value of α agrees with the experimental value within the limits of the statistical calculation error (Table 4).

The above investigations indicate that the planned reactor characteristics can be realized with the chosen reactivity modulator.

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PROGRESS IN ISOTOPIC POWER ENGINEERING IN THE USSR

G. F. Fradkin, N. E. Brezhneva,
Z. V. Ershova, N. I. Bogdanov,*
V. M. Kodyukov, A. N. Voronin,
A. G. Kozlov, Yu. A. Malykh,
B. V. Nikipelov, A. I. Ragozinskii,
V. V. Fedorov, and Yu. V. Chushkin

This report considers the design of isotopic thermoelectric generators for oceanographic and navigation devices; hydrographic, automatic radiometeorological, and magnetic monitoring stations; high-altitude cosmic ray stations, and other terrestrial research stations and devices.

The scientific and engineering aspects of the design of such power sources are discussed and the characteristics of some specific generators are listed.

Radioisotopic Fuel. More than 1000 radioactive isotopes are known at present. Successful selection of thermoelectric fuel requires a detailed study of the physical properties of the isotopes and in particular of the rate of change of the number of parent nuclei (burn-up rate) and the amount of energy released in one disintegration event. Most suitable as fuel are isotopes with a half-life in the range 100 days to 100 years (about 50 isotopes) of which 12-15 can be produced in large quantities.

The basic advantage of α-radioactive isotopes is the high energy release per disintegration most of which is converted into the kinetic energy of an α-particle and recoil nucleus whose range lies within the bulk of the radioactive material and the fuel can walls. In some radioactive isotopes (precursors of the radioactive transformation chain Ac^{227}, Th^{228}, and U^{232}) the total energy release per disintegration reaches up to 30-40 MeV, which is comparable to the fission energy of heavy nuclei. It is also important that in α-decay most transitions take place on the ground level of the daughter isotope and that in cases when the probability of transition on an excited level is appreciable, the excitation energy is quite low. β-Decay (in which, as a rule, the average energy per disintegration $\bar{E}\beta < 1$ MeV) is accompanied by bremsstrahlung and γ-radiation; on the other hand, however, β-radiative isotopes are more readily available.

The radiophysical properties of isotopes, as well as their sources and production methods, can serve as a basis for selecting isotopes most suitable as sources of electrical energy (see Table 1).

Radioactive fission isotopes and the most widely used Sr^{90} isotope are obtained principally by processing radioactive waste solutions. The group concentration method, based on calcium oxalate precipitation, is used for the production of radiochemically pure elements including Sr^{90}. The above method makes it possible to divide all radioactive fission elements into two groups: one including strontium, yttrium, the rare-earth elements, and americium, and the other including cesium, ruthenium, zirconium, niobium, and ballast impurities.

Most promising at present seems to be extractive separation of alkaline-earth elements with the liberation of pure strontium. This involves the use of the following extractants: a solution of di-2-ethylhexyl-orthophosphoric acid in kerosene in nitric acid medium and a solution of salicylaldoxime in tributylphosphate in alkali (sodium hydroxide or ammonia) medium.

*Deceased.

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TABLE 1. Comparative Characteristics of Some Isotopes Suitable for Thermoelectric Generators

Isotope	Half-life, yrs	Primary radiation	Total energy per one disintegration event, MeV	Density, g/cm³	Specific thermal power, W/g	Volumetric thermal power density, W/cm³	Activity per unit thermal power, Ci/W	Power		Principal production method
								γ-radiation, R/W·m²	neutron radiation, neutrons·sec/W	
Po^{210}	0,378	α	5,401	9,32	144,0	1340,0	31,4	$1,75 \cdot 10^{-4}$	—	Irradiation Bi^{209}
Ac^{227} (equil.)	21,7	α	34,332	—	14,85	—	4,94	1,0	—	Irradiation Ra^{226}
Th^{228} (equil.)	1,9	α	34,746	11,7	170,05	1995,0	4,875	3,0	—	Irradiation Ra^{226}
U^{232} (equil.)	73,6	α	40,174	19,05	5.0	95,0	4,2	3,56	—	Irradiation Pa^{231}
Pu^{238}	86,4	α	5,59	16,0	0,58	9.25	30,3	$6,35 \cdot 10^{-1}$	$4,85 \cdot 10^{3}$	Irradiation Np^{237}
Cm^{242}	0,445	α	6,213	13,5	122,5	1652,0	27,6	$6,7 \cdot 10^{-4}$	$1,62 \cdot 10^{5}$	Irradiation Am^{241}
Cm^{244}	17.9	α	5,895	13,5	2,89	39,0	28,6	$3,24 \cdot 10^{-4}$	$3,76 \cdot 10^{6}$	Irradiation of Am^{243} or heavy isotopes of plutonium
Sr^{90}	27,7	β	1,1	2,6	0.936	2,44	154,0	Bremsstrahlung	—	Fission products
Cs^{137}	29.68	β, γ	0,736	1,873	0,411	0,77	215,0	66,2	—	The same
Ce^{144}	0,78	β, γ	1,409	6,9	26,7	184,4	120,0	2,0	—	" "
Pm^{147}	2,62	β	0,062	—	0,338	—	2725,0	Bremsstrahlung	—	" "
Co^{60}	5,27	β, γ	2,607	8,71	17,5	152,0	65,0	84.5	—	Irradiation Co^{59}
Tu^{170}	0,354	β	0,321	9,3	11,35	105,5	525,0	Bremsstrahlung	—	Irradiation Tu^{169}
Ir^{192}	0,204	β, γ	1,1	22,5	59,7	1345,0	154,0	76,6	—	Irradiation Ir^{191}

One of the most important factors, in particular for short-lived isotopes, in the selection of an isotope as a thermal power source is its cost. This cost is critically affected by the possibility of using fast extraction methods which reduce the disintegration losses in the course of production. For example, Po^{210} can be obtained by rapid extractive distillation from enriched bismuth with multiple reuse of the bismuth.

The preparation of thermally stable polonium compounds from the vapor phase ensured a high measure of safety in the production and operation of generators.

Radioactive isotopes are used as fuel in the form of special preparations sealed in hermetical cans. From the point of view of radiation safety they present a sealed source of heat energy with an activity up to several hundred kilocuries with a high mechanical, thermal, and corrosion stability.

Since radioactive isotopes present a potential biological danger the fuel capsules must meet stringent requirements. The radioactive preparations should be hard and noncrumbling, nonsublimable, practically insoluble in sea and fresh water, and must not react with air, water, or the capsule material which also should be highly radiation and heat resistant.

From the point of view of radiation characteristics, the preparation should contain a minimum quantity of impurity radioactive isotopes with hard neutron and γ-radiation; the stable isotopes that enter into the chemical compound or serve as the carrier should contain predominantly elements with a low (in the

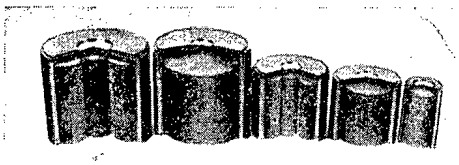

Fig. 1. Radioisotopic blocks based on Sr^{90}.

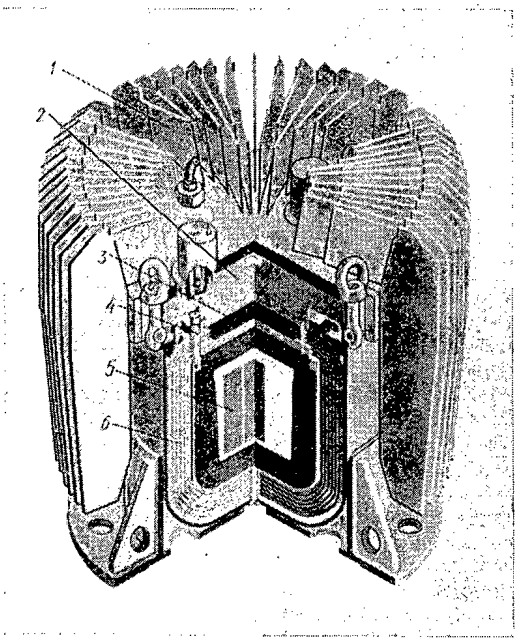

Fig. 2. Cross section of isotopic thermoelectric generator.

case of α-radioactive isotopes) or high Z (in the case of β-radioactive isotopes). The last requirement helps to reduce the yield of bremsstrahlung and neutron radiation of the isotopic assembly. The fuel material should have a sufficiently high thermal conductivity and not contain any appreciable amount of radioactive impurities with a half-life significantly different from that of the basic isotope. A low thermal conductivity causes a significant temperature drop inside the thermal unit and can impair its thermal stability; a high content of short-lived isotopes can cause a significant drop of initial power, whereas a considerable amount of long-lived isotopes is liable to reduce the specific activity.

The materials and construction of the capsule must ensure that the isotopic block remains intact and safe not only under normal operating conditions but also in emergency situations which can arise in the course of operation or transport and during storage (Fig. 1).

Presently used radioactive blocks have activities of the the order of tens and hundreds of kilocuries and are based on Ce^{144} (20,000 Ci), Sr^{90} (9,000-100,000 Ci), and Cs^{137} (50,000-150,000 Ci); Pu^{238}, Po^{210}, Cm^{242}, and Co^{60} radioactive blocks that meet all the above-mentioned requirements are also. The thermal power released by these blocks is in the 1 to 1000 W range.

Energy Release in Isotopic (Thermal) Blocks. Obviously, the total disintegration energy Q consists of the α-particle energy E_α, the recoil nuclei energy $(m_\alpha/m_{nuc}) \cdot \Sigma$, and (in the presence of fine structure) of the energy of emitted γ-quanta. In case of β-decay and spontaneous fission, one must also take into account the energy of fission products (E_{fp}), and the energy of β-particles and of γ-quanta:

$$Q = \left(1 + \frac{m_\alpha}{m_{nuc}}\right) \sum_i E_{\alpha_i} n_{\alpha_i} + \sum_i \bar{E}_{\beta_i} n_{\beta_i} + \sum h_{\nu_i} n_{\nu_i} + E_{fp}.$$

In general, the three-dimensional distribution of the energy absorbed in the isotopic block and the subsequent integration of this distribution cannot be calculated analytically but require the application of time-consuming numerical integration methods. However, considering that the range of α- and β-particles and of recoil nuclei is quite short (a fraction of a millimeter), practically all of their kinetic energy is absorbed within the fuel proper (the leakage bremsstrahlung is $\leq 1\%$). The energy absorbed within the radioactive block is equal to the total radioactive disintegration energy, while the specific heat release can be with a sufficient degree of accuracy assumed to be uniform over the entire block volume.

Since the range of γ-quanta is comparable with the block dimensions, calculation of the specific or total heat release due to them is one of the most difficult tasks. This causes a significant energy leakage in such isotopes as Cs^{137} and Co^{60}. Calculation of the specific heat release distribution resulting from the interaction of γ-radiation with matter at different points of the thermal unit requires numerical integration in several variables. Such an integration has been made for a Cs^{137} thermal unit in the form of an equivalent sphere.

The heat release has been checked experimetally with the aid of calorimetric devices designed to operate in hot chambers. The obtained results agree quite closely with theory.

Biological Protection. Adequate biological protection is of the utmost importance in the design and production of isotopic power sources containing kilocurie quantities of radioactive fuel. One of the particular features of this design is that on the one hand the dimensions of the radioactive block are comparable to the

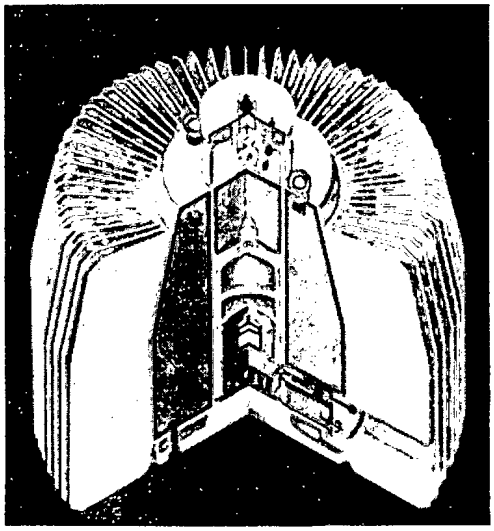

Fig. 3. Ce^{144}-based "Beta-1" thermoelectric generator.

Fig. 4. Sr^{90}-based "Beta-S" generator.

mean free path of γ-quanta in the fuel material (so that the effect of self-absorption in the source must be taken into account) and, on the other hand, the block dimensions are much smaller than the distance at which the dose rate P is limited to a safe value. In a point approximation we can thus assume that

$$P = \frac{Q}{R^2} F.$$

Here $F = \sum_i K_{ji} f_i e^{-\mu d} B(\mu d)$. In case of bremsstrahlung $F = f(I(E)/E)K_j(E)f(E)e^{-\mu d}B(\mu d)dE$, where K_j is the differential γ-constant, B is the buildup factor, f is the self-absorption coefficient, μ is the linear attenuation factor, and I(E) is the radiation intensity.

Conversion of Radioactive Disintegration Energy. One of the essential problems encountered in the design of isotopic power sources is the conversion of the radioactive disintegration energy into electrical energy. In the power range from one to several hundreds of watts the best results are obtained with the thermoelectric conversion method which offers high reliability, acceptable efficiency, and long working life.

Thermionic-emission and mechanical conversion methods are suitable for high-power generators (in the kilowatt range). Atomic batteries are efficient (in some cases) for electrical power sources in the micromilliwatt range.

The thermoelectric method is the most advanced one at present; semiconductor materials providing significant conversion efficiency (5-8%) are available for the low-temperature (up to 300°C), medium-temperature (300-700°C), and high-temperature (above 700°C) ranges. Combination of several different materials in cascade arrangements provides even now a conversion efficiency of 12-15% in experimental models.

Engineering design methods have been developed for single- and multi-cascade isotopic thermoelectric generators, and the radiation resistance of thermoelectric materials has been studied in neutron and γ-radiation fields. High concentration of recombination centers appreciably reduces the lifetime of electron-hole pairs, so that the effect of γ-quanta ionization on the properties of thermoelectric materials (α, σ, $\varkappa$, Z) is negligible. An estimation of the number of defects caused by prolonged (10^5 h) flow of neutrons (10^6-10^7 neutrons/cm^3 · sec) through such materials indicates that even without allowing for the effect of annealing the number of radiation defects is by two-three orders of magnitude less than the number of impurity defects.

Choice of Heat Circuit and Generator Design. An isotopic thermoelectric generator (Fig. 2) is a heat source consisting of a radioisotopic block 5, whose surface is partly covered by the thermoelectric converter elements 3 and structural joints 4. The rest of the surface is covered with heat insulation 6. The heat carriers are connected to the structural elements of the generator which through a system of coolers

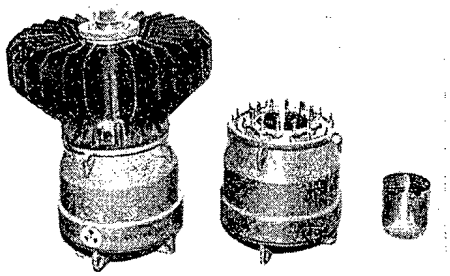

Fig. 5. Sr^{90}-based "Pingvin" generator.

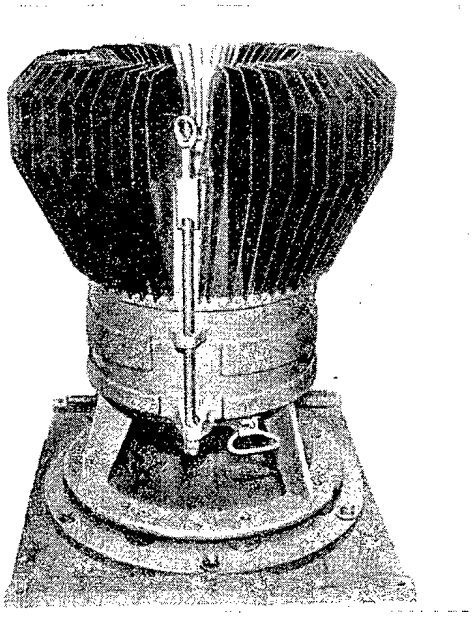

Fig. 6. Sr^{90}-based "Efir" generator.

1 dissipate the heat into the environment. From the point of view of the heat-exchange theory, an isotopic thermoelectric generator can be regarded as a system with an internal heat source whose thermal field is governed basically by thermal conduction or radiation inside the system and by convective (radiative) or contact heat exchange with the environment.

The principal problem in selecting and designing the heat circuit is the provision of maximum heat flow towards the thermoelectric converter so as to create on it a temperature drop corresponding to the optimum temperature range of the selected semiconductor material. Particular attention is given to the generation of a uniform heat flow through all converter elements and to the reduction of parasitic thermal resistances on its path. The abundance of factors affecting the thermophysical characteristics of the generator requires a detailed preliminary and experimental investigation.

Thermoelectric Generators with Ce^{144}, Cs^{137}, Sr^{90}, Pu^{238}, Cm^{242} (Po^{210}). The feasibility of using Ce^{144} and Sr^{90} isotopes in thermoelectric generators for powering automatic radiometeorological stations or other similar devices can be judged from an analysis of the properties of these isotopes and from a consideration of the state of their technology and production methods; it should be noted in this connection that Ce^{144} based generators can operate for one-half to one full year while Sr^{90} can serve for one to 10 years.

A consideration of the radiation yield and heat release in the isotopic block indicates that the principal contribution comes from the daughter isotopes Pr^{144} and Y^{90}; the chemical form of the fuel is $Ce_2 \cdot (MoO_4)_3$ and $SrTiO_3$. The thermoelectric converter is made of low-temperature alloys based on $Bi_2Te_3 + Bi_2Se_3$ and $Bi_2Se_3 + Sb_2Te_3$ which have the most favorable thermoelectric properties on the 200-600°K temperature range together with excellent reliability. The thermoelectric element system is connected as a single battery which considerably simplifies the generator assembly and battery replacement. Thermal insulation consists of a system of shields that ensure minimum thermal loss and enable the use of a quite simple heat removal control system together with the use of relatively short-lived Ce^{144} isotopes (Fig. 3).

During the years 1963-1970 Sr^{90} has been used in "Beta-2," "Beta-3," and "Beta-S" (Fig. 4) thermoelectric generators for powering various radiometeorological stations, in "Efir" and "Pingvin" generators for powering light and radio beacons and magnetic-monitoring stations, (Figs. 5 and 6), and in the dual-purpose "Angara" generator used for supplying both heat and electricity in high-altitude cosmic ray stations.

The use of the Cs^{137} isotope in electrical power supplies for various hydrographic stations is warranted chiefly by the fact that this isotope can be produced industrially in quantity, has an adequately long half-life, and by the possibility of using sea water as a biological shield. Among its shortcomings is the presence of accompanying γ-radiation (Cs^{137} at 660 keV and Cs^{134} at 1.367 MeV) and the relatively low

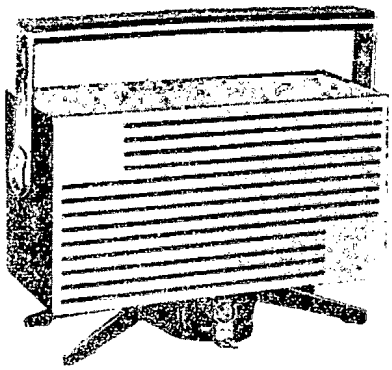

Fig. 7. Pu^{238}-based "MIG-67" generator.

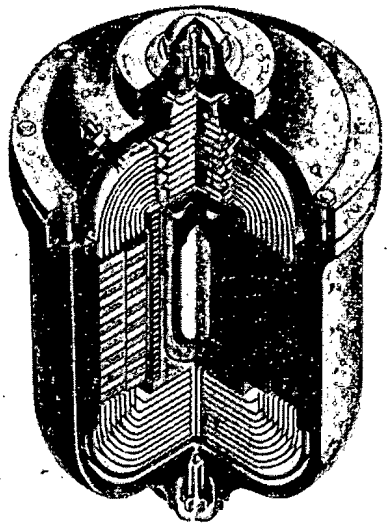

Fig. 8. Generator on cascade transformer.

specific release (0.123 W/cm^3). In all these devices the fuel is used in the form of cesium–lead–silicate glass as the least soluble cesium compound.

A distinctive feature of Cs^{137}-based fuel is that 70% of the radioactive disintegration energy is contained in the kinetic energy of γ-quanta; this makes it necessary to use massive heat blocks. By enclosing this block in a high-efficiency heat insulation system it is possible to direct maximum heat flow towards the thermoelectric converter and so to increase its power density by a factor of five to six.

The design of isotopic electrical power sources using Pu^{238} has been stimulated by the need of a portable general-purpose source.

The demands of minimum weight and size, as well as the desirability of low neutron and γ-radiation background, led to the use of fuels based on α-radioactive isotopes. The above demands are most fully met by Pu^{238} which, in fact, is the only α-radioactive isotope with a sufficiently long $T_{1/2}$ (86.4 years) and low accompanying radiation.

The design of type MIG-67 portable generators (Fig. 7) using Pu^{238} required the solution of several problems both in the design of the isotopic block and the micromodule thermoelectric converter. The small size of the generator causes increased heat loss through the structural elements of the generator. The loss of heat through the short mechanical joints is comparable to the loss through insulation so that is is preferable to locate the low-power isotopic (thermal) unit directly on the battery or between two batteries.

Unique properties of the Cm^{242} and Po^{210} isotopes (high specific energy release, relatively low γ-radiation intensity) as well as the potential feasibility of quantity production open up wide possibilities for the design of high-density power sources. The use of high-density radioactive fuel allowed the design of thermoelectric generators with cascade converters having an efficiency of 8-10% in the 300-850°K temperature range.

The positive experience gained in the design and operation of isotopic generators in various regions of the USSR and the expanding possibilities in quantity production of radioactive isotopes lead the way to further progress in isotopic power engineering in the USSR.

RESEARCH IN CONTROLLED THERMONUCLEAR FUSION IN THE USSR

L. A. Artsimovich

The problem of using nuclear fusion reactions for power purposes remains in the stage of preparing the scientific foundation on which the engineering developments of future thermonuclear fusion reactors must be based. This stage will end after methods for obtaining a new nuclear fuel based on deuterium or mixture with tritium have been developed. Such fuel must be a high-temperature plasma which satisfies two principal requirements: (1) the plasma temperature (or, more precisely, the temperature of its ionic component) must be of the order of $\sim 10^4$ eV (i.e., $\sim 10^8$ degrees); (2) the thermal enermy must be maintained in the plasma for a sufficiently long time. It is necessary that the condition $n\tau > A$ be satisfied, where n is the plasma concentration (the number of electrons per cubic centimeter); τ is the average time for which the energy is conserved (the ratio between the thermal energy reserve and the thermal flux from the plasma). The constant A depends on the composition of the nuclear fuel. For pure deuterium it is $\sim 1 \cdot 10^{16}$ under optimal assumptions, while for a mixture of deuterium and tritium (with equal amounts of the two components) it is approximately $1 \cdot 10^{14}$.

However, so far not a single experiment has been able to achieve the combined fulfillment of the conditions indicated (we are not speaking here of a thermonuclear explosion whose destructive energy is very difficult to use practically with a high enough efficiency). At the same time it may be stated that from year to year the physical parameters of the plasma are being improved.

Almost all the methods which used to create a high-temperature plasma are based on the idea of magnetic thermal insulation. The essence of the idea resides in the fact that the plasma must be immersed in a powerful magnetic field. The geometric shape of the field is chosen to be such that the electrons and ions of the plasma, which may move relatively freely only along the magnetic lines of force of the field, must not depart from a bounded region in space designed to be filled with plasma. In this case the transfer of energy to the chamber walls which confines the plasma may be (at least theoretically) reduced by many orders (i.e., the heat may be well trapped within the plasma).

Several versions exist for implementing the idea of magnetic thermal insulation. They form the basic competing trends in the problem of thermonuclear fusion. In recent years the scope of these trends has expanded as a consequence of the appearance of new propositions concerning ultrafast heating of matter by light and electron fluxes having a very high power. Under these conditions the use of magnetic thermal insulation may even turn out to be nonmandatory. At present one cannot yet make a final choice between the various approaches to the solution of the problem of controlled nuclear fusion or indicate the most promising direction in which the greatest effort should be concentrated. Therefore, in the next few years the program of scientific-research work must remain fairly broad. In the Soviet Union the principal elements of this program were defined fairly long ago, although the distribution of efforts toward their development is gradually changing.

In recent years the most intense development work in the USSR has been carried out in two directions: (1) the study of methods of obtaining a high-temperature plasma in closed (toroidal) magnetic systems; (2) the investigation of the properties of a high-temperature plasma in open magnetic systems having "magnetic mirrors."

Considerable attention was likewise devoted to an investigation of the interaction of a plasma with high-frequency electromagnetic fields. In addition, the study of processes which lead to the development

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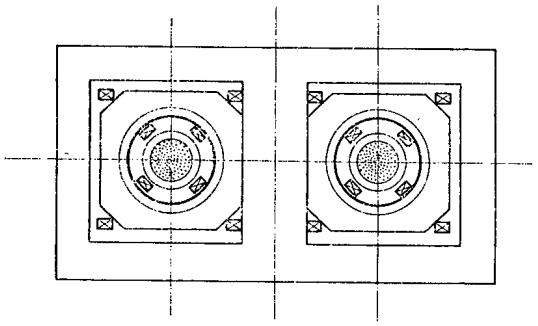

Fig. 1

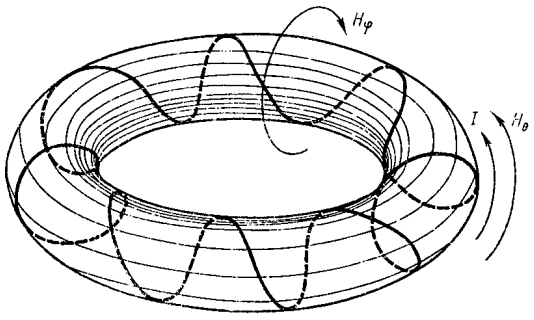

Fig. 2

Fig. 1. Diagram of a "Tokamak" installation.

Fig. 2. Structure of the magnetic field which confines the plasma in a "Tokamak" installation (the pitch of the helical line is greatly reduced for purposes of clarity).

of a so-called "plasma focus" – a very dense high-temperature plasmoid which is formed during the final stage of fast pinch of a plasma by the magnetic field of the current flowing through it – has been continued.

Comparatively recently a new trend in the thermonuclear fusion program appeared – the heating of matter to ultrahigh temperature by a laser pulse of very high power and such short duration that thermal insulation becomes meaningless during the heating phase.

Let us consider the basic results of investigations which have been performed according to the enumerated subdivisions of the overall program.

In closed magnetic systems a plasma may spread freely along the magnetic lines of the field which do not depart from a bounded region in space. It is formed inside a toroidal chamber and takes the form of an annular turn. One of the basic versions of the construction of such systems is the group of "Tokamak" installations. In installations of this kind the high-temperature plasma exists under quasisteady conditions. A current generated by the induction method flows along the annular plasma turn. The magnetic field of this current plays the principal role in ensuring the magnetic thermal insulation of the plasma. The heating of the plasma takes place at the expense of Joule heat released by the current.

In order to suppress destructive instabilities which are inherent in a plasma conductor with a powerful current flowing along it, a longitudinal external magnetic field is used whose lines of force are parallel to the current. Its intensity H_θ must exceed the intensity of the current field H_φ manyfold. The longitudinal field is generated by means of coils which are situated on the surface of a toroidal chamber (Fig. 1). The magnetic field in the plasma turn is a superposition of two fields, and therefore its lines have a helical structure (Fig. 2). They are strongly elongated along the turn, since $H_\theta > H_\varphi$.

Figure 3 shows the largest of the currently active "Tokamak" installations – the T-4 – whose construction was completed this year. It is actually a modernization of the previous T-3 installation. In the T-4 the diameter of the toroidal chamber is equal to 1.8 m. The diameter of the cross section of the plasma turn may reach 36 cm. It is restricted by an iris which is mounted in the internal shell of the toroidal chamber – the "liner." The liner is welded from thin stainless steel bellows. The outer shell of the chamber is constructed from thick sheet copper. The maximum intensity of the longitudinal field created by the coils on the axial line of the plasma turn is ~50 kOe. For the indicated maximum value of H_θ the limiting current in the plasma at which its stability can be maintained must not exceed ~300 kA. In order to excite current in the plasma a core transformer is used which has a toroidal chamber placed over its core. The length of the current pulse may be varied within the limits of several hundred seconds.

In order to study the physical properties of the plasma various means of plasma diagnostics are used. The plasma concentration is measured by means of probing with short radio waves. In experiments on "Tokamak" installations it amounts to approximately $2 \cdot 10^{12}$-$6 \cdot 13^{13}$. More exact data on the electron temperature T_e are obtained by measuring the scattering of a laser beam in the plasma. Such measurements were first carried out on the T-3 installation in 1969 jointly by English and Soviet physicists. The ion temperature T_i may be determined as a result of analyzing the energy spectrum of neutral hydrogen (or deuterium) atoms emitted by the plasma due to the charge exchange processes. The other method of determining T_i in deuterium plasma is based on recording neutron radiation which is of thermonuclear origin in "Tokamak" installations.

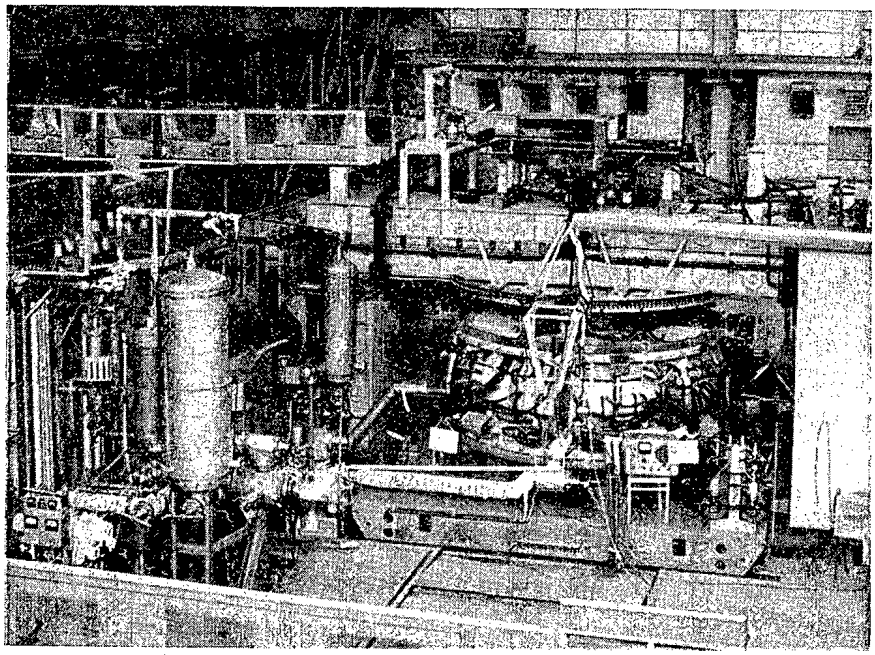

Fig. 3. Overall view of the T-4 installation.

At the I. V. Kurchatov Atomic Energy Institute investigations of the properties of a plasma in "Tokamak" installations have been in progress for longer than 10 years. In this institute the theoretical and experimental investigations are carried out in parallel. The choice of a specific experimental program was facilitated by the fact that at the very earliest stage of the research Shafranov developed the theory of equilibrium of a plasma turn and established the principal rule for magnetohydrodynamic stability of a plasma (the so-called Kruskal–Shafranov criterion). According to this rule it is necessary for the condition $(H_\theta/H_\varphi(a))(a/R)$ to be fulfilled for stability of the turn; here $H_\varphi(a)$ is the field intensity of the current on the plasma boundary; a is the radius of the turn cross section; R is the major radius of the toroidal system. The quantity q is called the stability margin. It depends on the distribution of the current over the cross section of the turn and for a bell-shaped distribution it must lie in the range from two to three.

A powerful impetus was given to the development of experimental research in recent years by the development of the "neoclassical" theory of diffusing and thermal conductivity phenomena in toroidal plasma systems. In this theory, whose foundations were laid by Sagdeev and Galeev, one important feature in the motion of charged particles in helical toroidal magnetic fields is considered as a consequence of which abrupt qualitative differentiation develops between the properties of a plasma turn and a straight plasma cylinder. In a helical toroidal configuration so-called "imprisoned" particles exist which oscillate along the lines of force and are reflected from those regions of the field where the field intensity is highest. The presence of imprisoned particles which may deviate noticeably from the lines of force during their motion must lead to an abrupt increase in the diffusion and thermal conductivity coefficients. In a plasma having a high temperature and low concentration the diffusion and thermal losses in "Tokamak" installation must increase approximately in the ratio H_θ^2/H_φ^2 in comparison with the value yielded by the initial version of classical theory. For "Tokamak" installations this means an increase in the rates of diffusion and thermal losses by a factor of several hundred. The appearance of the new theory allowed abrupt reduction of the discrepancy which existed previously between the experimental data and the theoretical calculations. However, as we shall see further on, this still does not mean that complete clarity has been achieved in the understanding of the mechanism of the processes which cause energy losses from a plasma in "Tokamak" installations.

In many years of experimental work in the "Tokamak" program a number of improvements were introduced in the construction of the installation. The improvement of the vacuum conditions under which the formation and heating of the plasma turn takes place allowed considerable reduction of the impurity content in the plasma. Correction of the geometry of the magnetic field and development of methods for controlling the position of the plasma turn in the chamber provided the possibility of abruptly attenuating the interaction of the plasma with the liner walls and the iris, as a result of which the temperature rose and the plasma

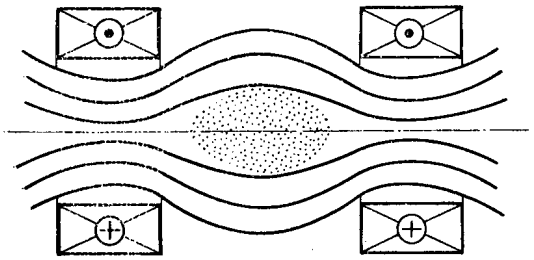

Fig. 4

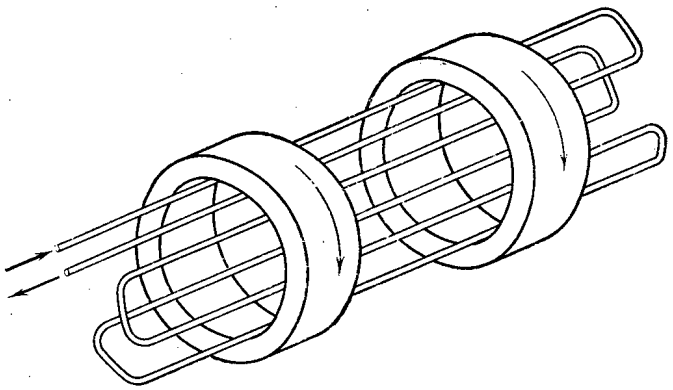

Fig. 5

Fig. 4. Diagram of the simplest magnetic trap.

Fig. 5. Shape of the winding which creates a magnetic field which increases on all sides of the plasma.

stability increased. The search for optimal operating modes of the process likewise facilitated an improvement in the principal physical parameters of the plasma.

Let us present the most important results of the experimental investigations which have been carried out according to the "Tokamak" program. The measurements of the thermal energy of the plasma which were carried out on the T-3 installation by various independent methods showed that in a steady-state thermal mode the ratio between the average magnitude of the gaskinetic plasma pressure $p = nk(T_e + T_i)$ and the pressure of the magnetic field of the current on the boundary of the turn $H_\varphi^2(a)/8\pi$ varies only slightly over fairly wide limits of variation of the initial conditions of the process. For a concentration which is not too small the ratio $\beta_\varphi = 8\pi\bar{p}/\bar{H}_\varphi^2(a)$ is 0.4 ± 0.1. This result may likewise be written in the following form*:

$$2Nk(\bar{T}_e + \bar{T}_i) \approx 4 \cdot 10^{-3} I_A^2,$$

where N is the total number of electrons per centimeter of length of the plasma turn; $\bar{T}_e$ and $\bar{T}_i$ are the average values of T_e and T_i (per electron or ion of the plasma); the magnitude of the current is expressed in amperes here. The electron temperature is usually considerably in excess of the ion temperature. The maximum value of T_e on the axis of the plasma turn in the T-3 installation reached $\sim 2 \cdot 10^7$ degrees† for $n_e \approx 2 \cdot 10^{13}$.

The profile of the temperature distribution over the cross section of the plasma turn may be approximated well by the function $T_e = T_e(0)(1 - r^4/a^4)^2$. A characteristic feature of this distribution is the flat top which is evidence of the fact that the coefficient of thermal conductivity reaches very large values near the axial line of the turn (where $H_\varphi \to 0$).

The processing of the results of measurements of T_i shows that the quantity T on the axial line of the plasma turn may be represented in the form of the following function of the process parameters:

$$T_i \approx 6 \cdot 10^{-3} \sqrt{I H_0 R^2 \bar{n}_e} \frac{1}{\sqrt{A}},$$

where $\bar{n}_e$ is the average magnitude of the electron concentration; A is the atomic weight of the substance.

The indicated dependence is in good agreement with predictions of the neoclassical theory of thermal processes in a plasma. However, since it was obtained by processing experimental data associated with a restricted range of the principal parameters, it is still early to speak of the proposition that the quantitative conclusions of neoclassical theory have been experimentally verified on this point. The maximum magnitude of T_i measured in experiments carried out on the T-3 installations was $\sim 7 \cdot 10^6$ for hydrogen and $\sim 5 \cdot 10^6$ for deuterium (in this case a noticeable neutron radiation was observed, which had a quasi-steady character and continued for 20-30 msec). The mechanism of the processes which determine the

*k is the Boltzmann constant which is equal to $1.37 \cdot 10^{-16}$.

†The temperature is frequently also expressed in electron-volts by means of the relationship: 1 eV is equivalent to 11,600 degrees. Henceforth we shall use both temperatures scales.

heat losses from the plasma turn has not yet been finally established. Evidently, the electronic component of the plasma loses energy considerably faster than is prescribed by the neoclassical theory. The conservation time of thermal energy τ_E in the plasma for the T-3 installation was 3 to 10 msec. When the initial experimental conditions were changed, τ_E changed in proportion to the skin-effect time. The relationship $\tau \approx 1.1a^2\bar{\sigma}_E$ holds, where σ_E is the electrical conductivity of the plasma averaged over the cross section.

The electrical resistivity of the plasma in "Tokamak" installations under ordinary experimental conditions exceeds the magnitude predicted by theory for hydrogen plasma under steady-state conditions severalfold. The cause of this anomaly has not yet been established. The increase in resistivity may be caused by the presence of ions of heavy impurities in the plasma. Moreover, it should be expected that for a low plasma concentration the passage of current must be accompanied by buildup of oscillations in the plasma, which leads to intensified retardation of the electrons.

One may speak of further prospects for research according to the "Tokamak" program only on the basis of a rather near extrapolation of the experimental relationships that have been found. If we continue to use Joule heating of plasma, then it will evidently be possible to advance to ion temperatures lying in the $(2\text{-}3) \cdot 10^7$ (2-3 keV) range and to bring up to the plasma concentration to $\sim 10^{14}$ and the duration for which the thermal energy is retained to tenths of a second if the geometric dimensions of the installations and the intensity of the longitudinal field are increased further. However, in order to reach a T_i level equal to 10^8 we shall require new more efficient methods of plasma heating. For this purpose one may in principle use injections of intense streams of fast neutral atoms into the plasma and resonance absorption of electromagnetic waves at plasma frequencies. One should also have in mind the possibility of using the method of "magnetic pumping" of energy by periodic contraction and expansion of the plasma; this may be achieved by applying an alternating control field $H_\perp$ directed parallel to the principal axis of the toroidal system to the plasma turn. The possibility of using so-called turbulent heating of a plasma under short pulses of a high induced voltage is likewise of interest. None of the methods enumerated has yet been tried on "Tokamak" installations.

A different version of closed systems is implemented in stellarators. In these devices the function of magnetic thermal insulation of the plasma current is assigned to a longitudinal field of external origin which must have a special fairly complex structure in order to carry out this function. The presence of so-called "rotational transformation" is characteristic of this structure; this transformation resides in the fact that the magnetic lines of force are rotated about the ring axis of the plasma current (if one travels along the chamber). The magnetic field in "Tokamak" installations has the same property, only in them the helical structure of the lines of force develop due to superposition of the fields created by the plasma current and the external coils. The presence of a ring current in the plasma is not mandatory for a stellarator. Therefore, in such a system the plasma may be created not only by means of an electrodeless induction discharge, but likewise by ionizing the gas by means of a high-frequency electric field caused by injecting a plasma jet from the outside.

In the USSR the development of methods for obtaining a high-temperature plasma in stellarators is being carried out at the Khar'kov Physicotechnical Institute and the P. N. Lebedev Physics Institute. Many stellarator installations have been constructed. The largest of them – "Uragan" – was constructed in Khar'kov. The length of the plasma turn of the installation was 10 m, while the maximum magnetic field intensity is 10 kG. Experiments have shown that one may obtain a plasma having high physical parameters in a stellarator only in the case in which the geometry of the magnetic field satisfies very rigid requirements and the field intensity is very high. At present we do not yet have an installation which fully satisfies these requirements. Therefore, most of the experimental information obtained in recent years is associated with conditions under which the plasma has a low density and a fairly low temperature. Under these conditions it was clarified that the decay rate of the plasma decreases with increasing angle of rotational transformation of the magnetic lines of force (i.e., with increasing steepness of the helical structure). The time for which the energy is retained in the plasma increases with increasing electron temperature. In the "Uragan" installation it reaches 0.5 msec for $n_e \approx 2 \cdot 10^{13}$. The value of T_e measured from laser-light scattering amounted to 100 to 400 eV. It should be noted that if the data on the diffusion velocities and thermal losses obtained from the "Uragan" stellarator are extrapolated and recalculated for the dimensions and magnetic field intensity of large "Tokamak" installations, values close to those which are characteristic of T-3 installations are obtained for the energy conservation time and the lifetime of the particles. At the present stage of development the stellarator program remains promising.

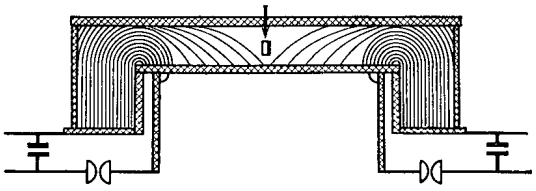

Fig. 6. Diagram of the installation for obtaining a "plasma focus."

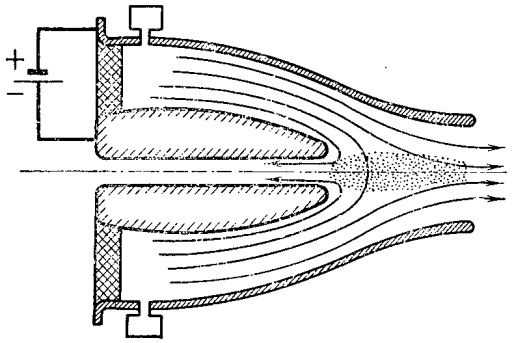

Fig. 7. Diagram of a "plasma compressor."

A traditional part of the overall program of work on controlled fusion in the USSR is the research on the properties of a hot plasma in open magnetic systems with magnetic mirrors. They are usually called mirror magnetic traps. In systems of this type fast charged plasma particles are confined by reflection from regions having an intensified magnetic field which are created on the boundary of the region occupied by the plasma. The simplest mirror trap is depicted in Fig. 4. The magnetic field is produced in it by two coils which carry current in the same direction. The electrons and ions of the plasma oscillate along the lines of force between regions having an intensified field which are situated in the regions in which the magnetic lines of force bunch together. However, based on a theoretical analysis one may conclude that the plasma must flow out rapidly from such a trap. This occurs because the magnetic field intensity increases along the lines of force in both directions from the middle region occupied by the plasma and simultaneously decreases in the radial directions.

A plasma is a diamagnetic substance, and therefore it tends to move in the direction of a decreasing field (i.e., it spreads out laterally). Experiments confirmed this theoretical prediction. Therefore, simple magnetic traps were replaced by more complex ones in which the magnetic field satisfied the so-called principle of the minimum H (i.e., it increases in all directions from the region occupied by the plasma). The first mirror trap satisfying this condition was constructed in the I. V. Kurchatov Atomic Energy Institute in 1961 (Fig. 5). In it the source of the magnetic field consisted not only of coils but also of six straight current-carrying conductors. The currents flow in opposite directions in the neighboring conductors. This system produces a field whose intensity increases on all sides of the central region. The very first observations of the behavior of a plasma in a trap having such a "hybrid" field revealed that all attributes of the most dangerous instability (so-called flute or diamagnetic instability) vanish in this field; this instability was previously one of the shortcomings of a system having magnetic mirrors. Because of this it was possible to increase the duration of particle and energy retention in the plasma by more than two orders. Thereafter several similar installations (PR-5, "Ogra-II," PR-6, PR-7) were constructed at the Atomic Energy Institute. In 1968 a plasma having an ion temperature of approximately $4 \cdot 10^7$ degrees (4 keV), a concentration 10^9-10^{10}, and an energy retention time of ~50 msec had already been obtained on the PR-5 installation.

The advance toward higher concentrations took place, in the face of fairly gloomy theoretical predictions. From the theory it follows that, having eliminated the coarsest form of instability, experimentors would encounter more refined ones – so-called "cone" instabilities – in attempting a further increase in plasma density; such instabilities develop in mirror traps as a consequence of the nonequilibrium distribution of the ions in the velocity space (this statistical nonuniformity is inherent in the very principle of the construction of such traps). According to theoretical calculations, if the longitudinal dimension of the region occupied by the plasma exceeds ~100 Larmor ion radii or if the transverse dimensions of the plasma are considerably smaller than the value indicated, then due to buildup of cone instabilities the plasma must be ejected very rapidly from the trap.

However, experiments did not substantiate these fears. A plasma having a concentration of 10^{11}, an ion temperature $1 \cdot 10^7$ degrees (1 keV) and a retention time of 15 to 20 msec was obtained on the PR-6 installation in 1968-1969; here the retention time was determined by processes of charge exchange of fast ions with atoms and bore no relation to any instabilities. In the experiments indicated the concentration threshold at which one of the forms of cone instabilities should have occurred was exceeded by an order.

Thereafter, the same installation was used to carry out investigations on the confinement of a plasma having an initial concentration $2 \cdot 10^{12}$ and an ion temperature $(1-2) \cdot 10^6$ degrees (100-150 eV). The study of a plasma having such a comparatively low temperature is of interest, since it simulates most completely the state of a plasma having thermonuclear parameters. Due to the relatively low temperature the dominant role in the plasma processes is played by Coulomb collisions between particles in this case (i.e., by the principal process which must determine the lifetime of the plasma in the traps in the absence of instability).

Measurements of the decay rate of this plasma showed that the initial stage of decay lasting 0.2 to 0.3 msec takes place without noticeable attributes of instability. The measurement of the velocity with which the plasma flows out of the trap during this initial state is in agreement with the results of calculations according to the classical theory for plasma stability (in such calculations only the leakage of particles through the magnetic mirrors as a result of Coulomb collisions is considered).

In 1967-1968 the "Ogra-II" installation was used to perform experiments for the first time on stabilization of various forms of plasma instabilities by means of an electric field which acts on the plasma via a system of feedbacks. In experiments with a plasma which did not have a very high concentration this method turned out to be very efficient. Then it was used successfully in joint experiments performed by Soviet and English physicists on the "Phoenix" installation at the Culham laboratory. Further experiments must clarify the problem of the possibility of using systems with feedback to suppress flute instability in simple mirror traps at a high plasma density.

One of the basic problems in the program of experimental research carried out with mirror magnetic traps resides in developing methods for filling traps with a dense plasma having a high ion temperature. The most natural version is the injection of an intense stream of fast neutral atoms which may be ionized in some way when they land in the trap. In connection with the research carried out on the "Ogra-I" and "Ogra-II" installations several fast-atom injectors were developed and constructed. The last of them had an equivalent current of 0.5 A at an energy of 20 keV and was designed for the LIN-5 installation which is under construction and has small geometric dimensions (the size of the region filled with plasma is ~5 liters).

After a certain reconstruction of the "Ogra-II" installation it was used to conduct experiments for the purpose of clarifying the possibility of storing a dense high-temperature plasma by means of so-called "turbulent heating." This is achieved as follows: using a special injector, a broad jet of cold plasma is produced which propagates along the system axis. A high-voltage pulse is applied between the plasma injector situated in one end of the trap and an electrode which is situated at its opposite end; the pulse lasts for about several microseconds. Under these conditions a longitudinal pulse current of the order of 20 kA develops in the plasma jet. As a consequence of the development of instability during the passage of a high-density current through a rarefied plasma, an intense turbulent heating of the electrons and ions takes place. The measurements of the plasma parameters which were conducted directly after the end of the heating phase showed that for a concentration of $\sim 1 \cdot 10^{13}$ the energy distribution of the ions covers a very wide interval, and the average energy of the ions is $\sim 1 \cdot 10^3$ eV. Using the AS installation, investigations of ion-cyclotron instabilities were carried out; such instabilities develop in a plasma having a very low density. An analysis of these instabilities may be of essential significance in evaluating various projects for recuperating the energy of particles which depart from a plasma through magnetic mirrors. It is not excluded that in these devices analogous instabilities may develop, as a consequence of which the efficiency of energy recuperation will decrease abruptly.

For many years considerable attention has been devoted to an analysis of the processes of interaction of a plasma with high-frequency electromagnetic fields. Such processes are of interest from two points of view. First, high-frequency fields may serve as a means for suppressing various instabilities which are inherent in a plasma which is confined in a magnetic field. In this case they are called upon to carry out an important auxiliary function in effecting magnetic thermal insulation. Second, the use of high-frequency fields for heating a plasma, and especially for heating the ion component in closed systems of the "Tokamak" type or in traps having magnetic mirrors, may have very great significance. Here one should distinguish between two fundamental versions: a) resonance heating using an external source of rf voltage; b) stochastic heating by fluctuating rf fields which develop as a result of the decay of ordered motion in a plasma (turbulent heating).

Experiments on plasma stabilization by rf fields (so-called dynamic stabilization) were carried out for several years at the Sukhumi Physicotechnical Institute, the I. V. Kurchatov Atomic Energy Institute,

and the Leningrad Institute of Electrophysical Equipment. The stabilization object in these experiments was chosen to be a plasma pinch having a powerful longitudinal current. In itself this object is magneto-hydrodynamically unstable. In a number of experiments it was shown that the principal instabilities may be suppressed if an additional longitudinal rf current is created in the plasma pinch.

Efficient stabilization of deformations of a plasma pinch can likewise be achieved in the case in which an rf field is generated by means of a system of four straight conductors situated parallel to the plasma pinch. Notwithstanding these apparently encouraging results, further development of the method of dynamic stabilization so far has not inspired great optimism. In making the transition to a dense high-temperature plasma, stabilization may be achieved only as a result of a very great increase in the power of the rf field.

In the immediate future this is practically unobtainable. Evidently, considerably better chances for success are offered by the use of rf fields for heating a plasma. Resonance heating of the ionic component of a plasma by an rf field at the Larmor frequency (ion cyclotron resonance) was studied at the Khar'kov Physicotechnical Institute. In this work the problem of the generation and propagation of ion-cyclotron waves in a plasma having a toroidal configuration was solved.

At the Atomic Energy Institute and the Leningrad Physicotechnical Institute several different methods for resonance heating of a plasma by rf fields are being developed: using electron cyclotron resonance, and so-called "hybrid" resonance. In addition, the heating of a plasma in toroidal systems at magneto-sound resonance is being studied at the Atomic Energy Institute; for this kind of heating the wavelength of the rf radiation is of the order of the transverse dimensions of the annular plasma turn.

In the Physics Institute, Academy of Sciences of the USSR a theoretical and experimental study is being carried out of the nonresonance method of heating electrons, which develops when the field intensity of the wave exceeds a certain threshold value. In the Institute of Physics Problems interesting experiments are being carried out on the heating of plasma by radio emission in the decimeter range.

In recent years the use of turbulent heating of a plasma by the current flowing through it has acquired an important role. This effect was detected for the first time in 1961 at the Atomic Institute in experiments on the heating of a plasma by magnetosound waves of high amplitude. As has already been indicated above, it was recently used successfully to heat the ionic component in the "Ogra-II" magnetic trap. At the Khar'-kov Physicotechnical Institute this method was used successfully to heat the ionic component of a plasma having a density $\sim 10^{15}$ up to a temperature of 2 keV. It is possible that in the presence of an anomalous resistivity the effect of turbulent heating will likewise be manifested in "Tokamak" installations, but in this case the electronic component is heated.

We should note briefly the fact that a high-temperature state of matter having a high density has been obtained by the formation of a "plasma focus." Here we encounter the limiting case of the utilization of magnetic thermal insulation, in which the duration of the sojourn of fast particles in the high-temperature zone of the plasma is very short but the density in this zone is very high. Figure 6 shows the successive phases of motion of the plasma layer which develops during a discharge in deuterium between two coaxial cylinders. The shock wave which develops in the plasma as a result of its rapid contraction undergoes cumulation near the axis. Under these conditions a plasma focus is formed having a diameter of the order of fractions of a millimeter and existing for tenths of a microsecond or a microsecond. The temperature in it is 1 to 2 keV for a plasma density of $\sim 10^{19}$. The pressure of the plasma in the zone of focus lies in the 10^4-10^5 atm range. Intense nuclear reactions occur at the plasma focus, and under these conditions the intensity of the neutron radiation in deuterium may reach 10^{18} neutrons/sec. The yield of neutrons increases approximately in proportion to the square of the energy supplied to the discharge as that energy increases.

The mechanism by which neutron radiation develops is not as yet clear, but it may be assumed that a considerable contribution to this radiation is made by the thermonuclear reaction in the hot zone of the plasma. In evaluating future prospects for this method of creating a high-temperature plasma it is necessary to focus attention on its principal shortcoming – the low energy efficiency of the heating process. If an energy of 10^5 J is supplied to the discharge, then the reserve of thermal energy which can be concentrated in the hot zone does not exceed several hundred joules. For such a low efficiency a hypothetical thermonuclear reactor based on the "plasma focus" principle could create excess energy only when conditions which are unattainable from the engineering standpoint are fulfilled (for this purpose an energy of the order of 10^9 J would have to be released in a time of $\sim 10^{-7}$ sec for each discharge pulse). In principle, however, it is not excluded that methods will be found in the future which will allow the situation to be altered

as a result of a cardinal improvement of the efficiency of the process. In this connection the idea advanced several years ago on obtaining a dense high-temperature plasma by "quasisteady" contraction of a plasma jet formed by means of a device similar to a coaxial plasma jet commands attention. In such a system, which has acquired the name of "plasma compressor," the plasma jet may theoretically be compressed without the formation of a shock wave. In this case the range of high density and high pressure may exist during a time interval such that the plasma configuration will not be disrupted by instability. An illustration of this process is shown schematically in Fig. 7. A conical plasma stream converges to the axis where it is converted into two jets. The motion of the plasma is maintained by electrodynamic forces which are created by the current flowing through the plasma between coaxial electrodes. It should be expected that the plasma will have the maximum density and maximum temperature on the axis near the neutral point 0. The plasma focus must be formed in this region. It may be postulated that the duration of the existence of this focus reaches 10 μsec in this system.

A completely new trend in the thermonuclear fusion research program has developed recently due to the rapid advance of laser technology. Using lasers one may now form light pulses having a length of the order of 10^{-9} sec (and even shorter ones) while concentrating large amounts of energy in them. By focusing such a light pulse onto a small surface sector of matter in the condensed (solid or liquid) state one may convert a piece of this substance into a plasma and raise its temperature to the thermonuclear level. Under these conditions (at least, in principle) the necessity of using magnetic thermal insulation is eliminated, since during a time of the order of 10^{-9} sec the thermal energy does not have time to depart from the heated volume. Experiments based on this idea were begun in the P. N. Lebedev Physics Institute several years ago, and in 1968 neutrons which had developed in a plasma heated by a laser beam were registered for the first time. The plasma was obtained as a result of focusing laser radiation onto the surface of lithium deuterides in a spot having a radius $<200\,\mu$. The radiation constituted a group of three-four light pulses $\sim 2 \cdot 10^{-11}$ sec long each, which were separated by intervals of $\sim 10^{-9}$ sec. The overall energy of a group of pulses was ~30 J, while the instantaneous power reached $\sim 10^{16}$ W/cm^2.

The maximum number of neutrons recorded in an individual experiment was $\sim 10^2$, which corresponds to an ion temperature of the plasma equal to 1-2 keV. Rough theoretical estimates show that in the given method of heating matter the total energy in a light pulse $\sim 10^{-9}$ sec long must amount to at least 10^5 J for 100% efficiency of the laser device itself in order to maintain the basic conditions which are required for a positive energy balance in thermonuclear reactions. Modern ruby and neodymium-glass lasers have an intrinsic efficiency of the order of a tenth of a percent. Under these conditions the energy of the minimal light pulse forming a reaction which is favorable from the energy standpoint must amount to at least several megajoules, while the total energy released in 10^{-9} sec must reach $\sim 10^9$ J (for a reaction in a mixture of deuterium and tritium). From the practical standpoint this means that the entire process (both in the laser system and in the equipment in which the pulsed heating of matter takes place) will have the character of a powerful explosion. From this it follows that the prospects of the laser method of generating thermonuclear reactions depends to a considerable degree on successes in developing lasers having a high efficiency.

Besides laser radiation, one may in principle likewise use intense beams of fast electrons for pulsed heating of matter in pulses of very short (nanosecond) length. Here the principal hopes are based on theoretical predictions that for a fairly high flux density of fast electrons (electrons having an energy of the order of several megavolts) the electron beam will convert condensed matter into a plasma. Under these conditions the processes of electron retardation in the substance completely alter their character, and collective interactions between the beam and the plasma will take first priority. Such interaction denotes buildup of intense electromagnetic oscillations in a plasma as a result of beam energy (i.e., the abrupt increase in the retardation of electrons with a corresponding reduction of their mean free path in the substance). The energy which the electrons lose will under these conditions go to heat the plasma. For a duration of electron irradiation of the substance lasting about 10^{-9} to 10^{-8} sec magnetic thermal insulation may turn out to be superfluous, just as it is for pulsed laser heating. The indicated method of heating may offer chances for practical application if the conclusions of the theory concerning anomalously large retardation of the electrons in ultradense plasma are confirmed, and a method of focusing beams with reduction of their diameter to ~1 mm for a current of $\sim 10^6$ is developed. Experiments which allow the possibility of the satisfaction of these conditions to be verified have only just begun.

A few words in conclusion. When the stated goal has not yet been reached one may evaluate progress in large-scale scientific-engineering development only if one can compare its states at different times.

At the time of the Third Geneva Conference we were able to obtain a plasma having an ion temperature close to 100-150 eV in closed magnetic traps, and the lifetime of the plasma did not exceed 1 to 2 msec. In the intervening period we have been able to elevate the ion temperature in "Tokamak" installations by a factor of 4-5 and to increase the energy retention time in the plasma by the same factor. In open traps we were able to create a stable plasma at a density of the order of 10^9-10^{10} in 1965.

At present we have been able to raise the density by two orders while retaining stability. These are the rates of progress along traditional paths. They are not so high as to provide a feeling of great satisfaction, but they are not so low as to serve as justification for pessimsm. Progress continues practically without a halt, although not as fast as we might like.

The ideas which have appeared in recent years on ultrafast heating of matter by powerful laser pulses and intense beams of fast electrons have evoked great interest in the problem of controlled thermonuclear fusion.

Evidently, in the nearest future we shall be able to judge the fruitfulness of these ideas on the basis of extensive experimental research.

BASIC PROBLEMS IN THE FLUORIDE METHOD OF REPROCESSING FAST-REACTOR FUEL ELEMENTS

Y. D. Veryatin, N. P. Galkin,
V. A. Zuev, V. I. Lomov,
V. P. Novoselov, V. N. Prusakov,
G. I. Chechetin, and Yu. D. Shishkov

Soviet nuclear power development plans call for the building of a large number of nuclear power generating stations in the coming decades, and fast-reactor power plants are prominent among them. Expected nuclear power development trends in the area of future fast reactors center around the use of UO_2 combined with PuO_2 as fuel (approximate ratio 20% PuO_2 to 80% UO_2).

The structural material for the fuel assemblies and fuel elements will be stainless steel, accounting for possibly 50% of the total weight of the fuel assembly [1]. As much as 10% of the fission products will accumulate in the fuel assembly of the fast reactor core during the reactor campaign. These characteristics combined impose specific requirements on the radiochemical technology associated with fast reactors.

Dry methods of uranium and plutonium recovery (nonaqueous methods) are being developed in the Soviet Union alongside improvements in aqueous techniques of spent fuel recovery. Greatest progress has been recorded in technological investigations of the fluoride method based on separation and purification of fuel components by utilizing the differences in the volatility of their fluoride compounds. This method offers some distinct advantages: the possibility of compact radiochemical production with intensification of the technological process; simpler ways of achieving nuclear safety conditions; immobilization of radioactive wastes in a solid compact form, etc.

A broad research program of the fluoride technology of nuclear fuel reprocessing has been undertaken in the Soviet Union. In addition to physicochemical research, the program includes experiments on "cold" and "hot" facilities with remote-control operations, development of equipment for decladding fuel elements, fluorination of fuel, separation of fuel components, study of radioactive wastes disposal technology, development of components for remote-control equipment, and so forth.

Dismantling Fuel Assemblies by Heat

The process of recovering nuclear fuel is necessarily preceded by treatment of the highly active fuel assemblies, opening them up and separating the fuel from the structural materials. This requires that at least some minimum quantity of wastes be obtained in a form suitable for long-term storage. The practical solution of these problems in the radiochemical industry by familiar methods runs into great difficulties, sometimes insuperable ones.

The solution of one of these problems is found in the development of a method employing application of heat to open up irradiated fuel assemblies, and involves melting down the structural materials and removing the fuel from them [2]. This heat application method makes it possible to treat the fuel assemblies by remote control, either as whole units or by zones, since separate recovery of active zones and shielded zones is advisable on occasions, and this is even more likely the case when massive parts of fuel assemblies containing no nuclear fuel are to be removed. The fuel assemblies are separated quite readily by the method used in cutting of steel tubing [3]. In induction heating of narrow segments of the outer jacket to temperatures close to the melting points of the structural materials, the jacket may rupture, so that the

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assembly is separated into zones, and the jacket is removed together with the head and the shank. In this approach, the liquid, gaseous, and powdery wastes inevitably accompanying the other methods are eliminated.

Calculations show that an irradiated fuel element from the core of a fast reactor will heat up to temperatures above 800°C in air. The method of opening up the fuel assembly by heat calls for the use of this self-heating of the assemblies, plus additional applied heating in vacuum or in an inert gas medium, till the assemblies fail, which will occur at the melting points of the structural materials (1400-1500°C).

Physicochemical investigations and calculations related to the study of the effect of the gaseous medium, the composition of the steels, the type of fuel and its physical state, the time, temperature, methods used to heat the assemblies, and various other factors, on the process provide the theoretical groundwork for the process of opening up fuel assemblies by heating them [2, 4]. These investigations imply that it would be possible to open up the fuel assemblies without letting them stand for a while first, and with practically complete separation of the structural materials from the fuel accompanied by minimum losses of the fuel. It has been found that the content of uranium (plutonium) in the steel separated from the rest of the fuel assembly ranges from 0.01 to 0.1%, and depends on the conditions and equipment used in the process. The process is distinguished by high productivity, with reliable nuclear safety safeguards, and elimination of liquid process wastes and the dust-laden fraction, while the quantity of gaseous and solid wastes can be minimized by the method described in [5, 6].

The heavy accumulation of fission products in irradiated nuclear fuel is accompanied by physical and structural changes in the fuel. The effect of these changes on the process of opening up the fuel assemblies by application of heat is determined in the first instance by the state of the fission products in oxide type nuclear fuel.

When the molecules of uranium dioxide and plutonium oxide break down, oxygen is liberated, and this oxygen then oxidizes the fission products and the remaining uranium dioxide under high-temperature conditions. All of the fission products can be separated into two distinct groups in terms of the free energy of formation of the oxides. The first group, capable of forming oxides more firmly bound than uranium dioxide, is made up of the rare earths, yttrium, plutonium, barium, strontium, and zirconium, all of which will be oxidized by the oxygen liberated in the process. It has been shown that some of the oxides are capable of interacting with uranium dioxide, while the oxides of zirconium and cerium form solid solutions with uranium dioxide. The second group includes those elements which form oxides less firmly bound than uranium dioxide (iodine, rubidium, cesium, ruthenium, palladium, niobium, technetium, and molybdenum). The elements exhibiting low vapor pressure (molybdenum, palladium, niobium, rhodium, ruthenium) are found in the elemental state, and interact with each other to form alloys [7, 8] in which uranium and plutonium can be detected as metals [9].

The remaining free oxygen can either interact with uranium dioxide to form nonstoichiometric oxide of composition $UO_{2.05}$, or may partially oxidize the elements belonging to the second group, for which oxide formation conditions are found to be favorable (this outcome will be favored by the appreciable mobility of the superstoichiometric oxide present in the uranium dioxide). For example, iodine, rubidium, or cesium, having a higher vapor pressure, will react to the high-temperature gradients by migrating toward the cladding in the range of relatively low temperatures where such chemical compounds as oxides or iodides of the alkali metals, etc., can form. Xenon and krypton are partially retained in the uranium dioxide lattice, but are for the most part liberated in the form of gases filling up the cavity of the fuel element. The quantity of free gases present depends on the intensity of the irradiation, on the temperature, and on the percentage burnup. It has been demonstrated experimentally that about 80% of the gaseous fission products present are liberated when the burnup of plutonium dioxide is 6.1%, and reaches a figure of 95% when the burnup of $U^{235}O_2$ is 10% [12].

The products of this method of breaking apart fuel assemblies by heat are the nuclear fuel, a metal ingot, substances capable of sublimation, gases, and aerosols. The behavior of some of the elements and their oxides in oxide fuel, at temperatures above the melting point of the steel, has been studied, as well as the distribution of those elements in the products left over by the process.* This investigation was conducted using uranium dioxide pellets into which additions simulating the fission products had been introduced. The amount of such additions was determined by calculating 5, 10, and 20% plutonium burnup. The results of the experiment were in excellent agreement with data obtained by opening up irradiated uranium oxide fuel elements with 1.5% burnup. The impurity distribution between products of the process of opening

*The experimental data were obtained by A. T. Ageenkov, G. B. Borisov, Yu. D. Dogaev, and others.

TABLE 1. Distribution of Elements in Products of Heat-Induced Breakup of Fuel Assemblies

Products of heat-induced breakup	Elements, wt.%													
	U, Pu	Fe, Ni, Cr	Co	Mn	I	Te	Cs, Rb	Sr, Ba	Y, rare earth	Nb, Mo	Zr	Rh, Pd	Tc	Xe, Kr
Fuel pellets.	99,948	0,7	0,7	0,6	95,0	95,0	99,5	99,90	99,94	94,95	98,99	95,0	95,0	5—20*
Steel ingot	0,050	99,2	99,3	89,4	—	—	0,3	0,05	0,05	5,00	1,00	5,0	5,0	—
Sublimates	0,002	0,1	—	10,0	5,0	5,0	0,2	0,05	0,01	0,05	0,01	—	—	—
Vapor phase	—	—	—	—	—	—	—	—	—	—	—	—	—	80—95

*Depending on percentage burnup.

up the spent fuel elements by application of heat was determined on the basis of those investigations (Table 1).

Nuclear Fuel. We realize from the data in Table 1 that fuel assemblies opened up by application of heat offer high recovery yields of uranium and plutonium (99.95%) plus separation of the structural elements from the fuel (over 99%). The chemical, physical, and structural changes taking place in the fuel under irradiation do not exert any pronounced effect on the degree of separation. Most of the fission products, except for krypton and xenon, remain practically entirely within the fuel.

Steel Ingot. The stainless steel contains slight quantities of uranium and plutonium (less than 0.05%), radioactive isotopes Mn^{54} and Co^{58} forming when the steel is irradiated, and as much as 2% fission products [13]. It is mainly those elements present in the fuel in their metallic state (rhodium, ruthenium, niobium, molybdenum, etc.), as well as slight amounts of zirconium and cesium, which find their way into the steel. This steel can be directed to long-term storage, with no additional processing.

Sublimates. In dismantling of steel assemblies by heat, a slight quantity of the fission products sublimates: 5.0% I and 0.2% Cs, out of the total quantity present. Most of the activity of the sublimates is due to the presence of Mn^{54}, the content of which increases with increasing duration of the heating process. No increase was observed in the quantities of the other isotopes, and the uranium content did not exceed 0.002%.

Gases. Xenon and krypton present in the free state are liberated immediately after the fuel elements fail. A portion of the gases accepted into the uranium dioxide crystal lattice remains almost entirely in the fuel. Uranium losses with the aerosols are virtually nonexistent.

Fluorination of Spent Fuel by Gaseous Fluorine and Halogen Fluorides

Declad fuel is directed to the recovery cycle in order to effect complete removal of fission products and separation of the uranium and plutonium, and conversion of the uranium and plutonium fluorides to oxides. The complete recovery cycle includes the following steps: fluorination of the fuel, sorptive purification of uranium hexafluoride, purification by rectification, thermal decomposition of the plutonium hexafluoride, and pyrohydrolysis of uranium fluorides and plutonium fluorides.

Purification of the fuel in fluorination is based on the differences in the vapor pressures of the fluorides forming during the process, and the different reactivities of the fluorides of uranium, plutonium, and the fission products.

Since some of the fission products (e.g., the rare earths, alkali earths, alkali metals, etc.) form practically nonvolatile fluorides with fluorine, they can be successfully isolated from the uranium and plutonium directly as the fuel is being fluorinated.

Investigations carried out on irradiated material have for the most part confirmed that possibility, but a few discrepancies are on record. For example, there are present in the vapor phase, in addition to the uranium hexafluoride, and elements forming volatile fluorides (ruthenium, niobium, tellurium), as well as zirconium, to a partial extent, also cesium, strontium, cerium, and other elements the vapor pressures of whose fluorides are well below the equilibrium vapor pressure at the fluorination temperature. The last-named elements go over into the vapor phase in the form of aerosols and dust. For that reason cermet filters with pore size 10-30 μ, made of sintered nickel powder, were inserted in the uranium hexafluoride stream to remove aerosols and dust at 150°C. Investigations disclosed that filters of this type are capable

of efficiently removing dust and aerosols from the uranium hexafluoride, and that the uranium hexafluoride downstream of the fluorination and filtration steps contains only those fission products whose volatility is close to the volatility of uranium hexafluoride. These elements are neptunium, technetium, tellurium, niobium, and ruthenium.

The study of the behavior of plutonium when uranium–plutonium fuel containing simulators of fission products is fluorinated, showed that the presence of certain elements (cesium, strontium, barium, and rare earths) contributes to plutonium retention in the fluorination residue, because of the formation of strong complexes with plutonium fluoride. At the same time, some of the fission products contribute to sublimation of the plutonium during fluorination, by displacing the plutonium from the complexes formed, it would appear. A definitive answer as to the quantity of plutonium present in the fluorination residues could be obtained only from representative tests carried out on irradiated real (not simulated) fuel.

Recently, attention has been centered on research on other fluorinating agents for spent fuel, halogen fluorides for instance. The heightened interest in applications of halogen fluorides as fluorinating agents stems from the possibility of separating uranium and plutonium at the "head" of processes designed to remove fission products from fuel, with the aid of those agents.

In the case of such halogen fluorides as chlorine monofluoride, chlorine trifluoride, bromine trifluoride, and bromine pentafluoride, which are used in reprocessing spent fuel, it is not only their selective responses to uranium that are of characteristic interest, but also the lower temperature of the process and the lesser degree of exothermicity in the fluorination reactor. The most promising fluorinating agents at this writing are bromine pentafluoride, which is heat-stable to 460°C, and bromine trifluoride.

Investigations of the interaction between gaseous bromine pentafluoride and gaseous bromine trifluoride with sintered pelletized uranium dioxide confirmed the feasibility of direct fluorination without prior oxidation to the uranyl–uranate (mixed oxide) at higher rates than is the case with fluorine.

Investigations of fluorination of a mixture of uranium dioxide and plutonium dioxide corresponding to the composition of the fuel in a fast reactor core provided confirmation of the selective sublication of uranium hexafluoride on the fuel mixture. As a result of fluorination at a temperature of 350°C, the uranium goes over completely into the vapor phase in the form of uranium hexafluoride, while the plutonium remains in the solid precipitate in the form of plutonium tetrafluoride.

Equipment for Fluorination of Spent Fuel

The presence of a large quantity of fission products in the irradiated fuel arriving at the fluorination step, and the high exothermicity of the fluorination reaction of uranium dioxide ($\Delta H^0_{298} = -259$ kcal/mole) impose strenuous requirements on the equipment needed to realize the process. The process equipment must be capable of bringing about rapid extraction of heat from the reaction zone, as required for carrying out the fluorination process under controlled temperature conditions. These requirements are satisfied by the fluidized bed process equipment being developed in the USSR and in other countries [14-17].

The fluorination process of uranium oxide in a fluidized bed reactor, with a bed of aluminum oxide and uranyl uranate particles, is under investigation in the Soviet Union. The fuel feed to the process, in the form of pellets freed from the fuel elements, is subjected to prior oxidation, and the particulates of the uranyl–uranate forming are then fluorinated by fluorine gas.

Various equipment combinations are being tried out for the process. Attention is being focused on the following topics: 1) the equipment linking the oxidation and fluorination processes; 2) the stability of the fluidized bed fluorination process and how it is affected by various process factors; 3) filtration of off-gases containing heavy dust loads.

Three possible variants of the process of reprocessing pelletized uranium dioxide in uranium hexafluoride are being investigated. In the first variant, the uranyl–uranate mixed oxide is isolated in one reactor and it is fluorinated in another. In the second variant these two processes are carried out in a single reactor but are separated by zones in that reactor, with the pellets oxidized in the bottom zone of the reactor and the uranyl–uranate mixed oxide forming when those particles are oxidized being fluorinated simultaneously in the top zone (similar to the process worked out at Argonne National Laboratory in the USA). In the third variant the oxidation and fluorination are also carried out in the same reactor, but the processes are separated in time: first the fuel is oxidized, and then it is fluorinated.

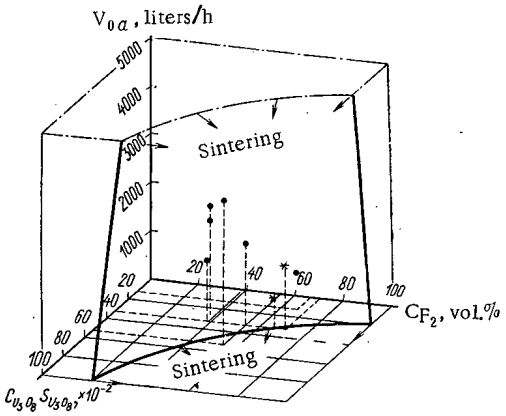

Fig. 1. Stable fluidization region: Data points indicate parameters at which the process is stable, while asterisks denote parameters at which sintering is observed.

At the present time, attention is being centered on the second and third variants of combining the two processes. The final selection of process equipment awaits the completion of tests on a pilot-plant test rig.

One of the key problems to be tackled is the danger that the uranium fluorination process in the fluidized bed might break down because of the sintering of the particles. This question is being investigated thoroughly at the present time. Some preliminary considerations on the existence of a region where fluorination of uranium oxide would proceed stably can be advanced on the basis of the results to date. These considerations are based on experimental observations which bear out the view that sintering takes place when the heat loads on the reaction zone are reasonably heavy.

The heat load on a volume element of the reaction zone with a fluidized bed is given by the formula

$$q = KC_{F_2}C_{U_3O_8}S_{U_3O_8},$$

where q is the amount of heat liberated, kcal/h $\cdot$ m^3; K is a constant representing the coefficient of mass transfer multiplied by the specific amount of heat liberated per mass fluorine, m $\cdot$kcal/h $\cdot$kg F_2; C_{F_2} is the fluorine concentration, kg/m^3; $C_{U_3O_8}$ is the U_3O_8 concentration in the fluidized bed, kg/m^3; $S_{U_3O_8}$ is the specific area of the U_3O_8, m^2/kg.

According to the condition for thermal balance,

$$KC_{F_2}C_{U_3O_8}S_{U_3O_8} = \alpha F_m \Delta T,$$

where K is the coefficient of heat transfer from a volume element of the fluidized bed to the reactor wall, kcal/m^2 $\cdot$ h $\cdot$ °C; F_m is the surface available for heat transfer on the reactor wall per volume element of the fluidized bed, m^2/m^3; ΔT is the temperature difference between a volume element of the fluidized bed T_b and the cold wall of the reactor T_w, °C.

Clearly, sintering in the volume element will occur whenever a certain temperature T_{max} is exceeded (we assume here that this temperature remains constant for the physicochemical system in question). The condition for maintenance of stable fluidization will assume the form:

$$\gamma C_{F_2}C_{U_3O_8}S_{U_3O_8} < T_{max} - T_w,$$

where $\gamma = K/\alpha F_m \approx$ const for the specific process, and the range of stable fluidization is described by the equation

$$C_{F_2}C_{U_3O_8}S_{U_3O_8} = \frac{T_{max} - T_w}{\gamma},$$

which is the equation of a hyperbola in the coordinates

$$C_{F_2} - C_{U_3O_8}S_{U_3O_8}.$$

When the flowrate of the fluidizing agent V_{0a} increases, α follows suit, and apparently so does T_{max} (this last fact meaning that sintering will begin at a higher temperature level because of the increased kinetic energy of the fluidized-bed particles). This means in turn that, as V_{0a} increases, the hyperbola will be shifted further away from the origin of coordinates.

Accordingly, in the system of coordinates $C_{F_2} - C_{U_3O_8}S_{U_3O_8} - V_{0a}$ the range of stable fluidization is characterized by a surface (see Fig. 1) which forms a hyperbola in any horizontal cross section and deviates slightly from the vertical axis V_{0a}. When the process is carried out with parameters lying within the volume bounded by this surface and by the coordinate planes, there will be no sintering of the fluidized bed. The diagram shows some preliminary results plotted on the basis of an experimental determination of the coordinates of the surface.

The filtration system is one of the basic parts of the process equipment. Investigations carried out with cermet filters showed a relatively rapid fall in their permeability, despite regeneration by pulsed

TABLE 2. Separation Coefficients of Binary Systems Obtained at 74°C

System	Microconstituent	Ideal separation coefficient	Experimental separation coefficient
$UF_6 - MoF_6$	MoF_6	1,73	1,22
$UF_6 - WF_6$	WF_6	2,86	1,39
$UF_6 - SbF_5$	SbF_5	26,5	6,53
$UF_6 - NbF_5$	NbF_5	660,9	217,6

scavenging blows. The fall-off in the permeability of the filters can necessitate frequent replacement of the filters, a solution which is not acceptable when the process is designed for remote control. Nevertheless, the first models of our reactors were equipped with 40 mm diameter cermet filters of pore size 10 μ. We carried out investigations on the design of continuous packed filters using aluminum oxide particles. These filters allow complete regeneration of filter permeability by periodically dumping the dust that has settled on the surface layer with an insignificant portion of the filter coat, and do not have to be replaced by remote control.

Taking the above considerations into account, we carried out appropriate tests on process equipment consisting of a gas distributor system, a reaction zone, a cooling zone for the off-gases, and a packed filter. This equipment was 130 mm in diameter, and was intended for experimental reprocessing of spent oxide fuel in separate batches weighing 30 kg.

High mass-transfer efficiency is achieved in the fluidized-bed reactor, with excellent conditions favoring removal of the heat liberated as a result of the exothermic fluorination reaction. But some problems in reactor design still require more careful work (unstable operating conditions, heavy entrainment of dust, consequent overloading of the filters, the need to perform prior oxidation of the fuel as a separate reaction in the process).

Fluorination of the packed layer of sintered fuel pellets directly has some attraction because of the simple reactor design, absence of dust carryover, and the stable behavior of the process. Investigations of the process of fluorination of sintered pelletized uranium dioxide in a fluidized-bed reactor showed that the principal regularities of the process are described by the kinetic equations of combustion of a carbon bed, but that there exists a concentration-dependent limit to the fluorine content in the original fluorinating mixture. This limit is 50% of the fluorine concentration in the original mixture. When the fluorine content is any higher, there will be overheating of the bed of pellets and accompanying sintering, leading to a breakdown in the process.

Absorption of fluorine takes place in a small zone of the bed of pellets. The height of the fluorination zone can be calculated by using the equation

$$x = \frac{V \ln \frac{C}{C_0}}{S_0 \beta},$$

where V is the rate of filtration of the gases; C_0 and C are the concentration of fluorine in the feed and at the height x; S_0 is the reaction surface; β is the mass-transfer coefficient.

The calculated height of the fluorination zone is 90-150 mm. A multizoned tubular type reactor design with special nuclear safety features was proposed and tested, as a tool for intensifying the process of fluorination in the moving bed, and for improving the reactor thermodynamic conditions. The reactor features zonal feed of the fluorinating agent, and automatically maintains a constant amount of material to be reprocessed within the reaction zone.

Tests on a three-zone moving-bed reactor showed that the throughput of the reactor was 28 kg UF_6 ($dm^2 \cdot h$) when the fluorine concentration was 20 vol. %.

Sorptive Purification of Uranium Hexafluoride

Extraction of fission products from fuel fluorination products can be handled by passing the gas stream through porous sorbents with a base of fluorides of alkali metals and alkali earths.

Of the fluorides of the alkali metals, the most promising for this application are LiF and NaF, and of the fluorides of the alkali earths the most promising are MgF_2, CaF_2, and BaF_2. All of these sorbents, with the sole exception of NaF, undergo virtually no interaction at all with uranium hexafluoride at low temperatures, but nevertheless form complexes with the fluorides of the fission products.

Sodium fluoride and magnesium fluoride were the principal choices for sorbents in our investigations. Sorptive purification demonstrated a very high efficiency. The following purification factors were obtained for the various elements: $Zr^{95} > 10^5$, $Ru^{106} > 10^5$, $Nb^{95} > 10^5$, $Cs^{137} = 10^7$, $Sr^{90} = 10^8$, and $Ce^{144} = 10^8$.

Purification of Uranium Hexafluoride by Rectification

Rectification provides an alternative method for purifying uranium hexafluoride [18, 19].

Experiments designed to study the phase equilibria of mixtures of uranium hexafluoride and higher-order fluorides of the heavy metals have yielded positive results. The systems investigated, UF_6-SbF_5, UF_6-RuF_5, and others, revealed excellent mutual solubility of the components. There is no basis for assuming any further complications in carrying out the process of purification of uranium hexafluoride by rectification.

Investigation of the static behavior of dilute solutions showed a rectilinear relationship between the equilibrium components of the vapor phase and liquid phase is typical of the range of impurity concentrations below 0.5 wt. %. The results obtained imply that the constant relationship persists all the way to $\sim 10^{11}$%.

Experimental data on the separation coefficients of the binary system UF_6-MoF_6, UF_6-WF_6, UF_6-SbF_5, UF_6-NbF_5 in the range of concentrations of MoF_5, WF_6, SbF_5, and NbF_5 extending from 10^{-1} to 10^{-6} wt. % (Table 2) confirm the possibility, in principle, of carrying out intensive purification of uranium hexafluoride and getting rid of fission products.

Investigations of the many-component system $UF_6-MoF_6-SbF_5-NbF_5$ in the range of vanishing concentrations of Mo, Sb, and Nb demonstrated that the trace components exert practically no effect whatever on each other, such as might affect their equilibrium distribution or the process of purification of uranium hexafluoride by rectification.

The height equivalent to a theoretical stage, a parameter characterizing the kinetics of the process of separating mixtures of fluorides with a uranium hexafluoride base, attains the following values: UF_6-MoF_6 2-5 cm; UF_6-WF_6 6-10 cm; UF_6-NbF_5 10-15 cm; UF_6-SbF_5 9-13 cm.

The UF_6 throughput of the rectification towers is ~ 1 g/cm$^2 \cdot$sec.

Practical verification of the results of the laboratory studies was achieved on pilot plane rectification towers. Uranium hexafluoride of spectroscopic grade purity was obtained with the most varied feed compositions. The work done with the pilot-scale rectification towers made it possible to master technically suitable approaches to their exploitation. Simultaneously, techniques for compressing the gaseous uranium hexafluoride were successfully worked out.

Separation of Uranium and Plutonium

One of the salient topics in the fluoride technology of recovery of spent fuel elements is separation of the uranium and plutonium. This separation can be effected by relying on the differential abilities of these elements to form higher-order volatile fluorides with fluorine, as well as on the different reactivities of the uranium hexafluoride and plutonium hexafluoride. Uranium dioxide becomes involved more readily in the interaction with fluorine, to form a higher-order fluoride, than does plutonium dioxide. The resulting uranium hexafluoride is more inert in its chemical activity than is plutonium hexafluoride, which acts as a more vigorous fluorinating agent than fluorine in fact, under certain conditions. Since many fission products capable of interacting with plutonium hexafluoride are contained in the spent fuel in the irradiation process, and lower-valency uranium compounds are also present, the plutonium hexafluoride is capable of oxidizing the latter, and becoming reduced to a tetrafluoride in the process.

The combination of these two differences in the properties of uranium and plutonium in a fluorine medium is what constitutes the prerequisite for separation of uranium and plutonium in the course of the fluorination process.

Experimental verification of the feasibility of separating uranium and plutonium in fluorination by means of a dilute inert gas (volume ratio 1:1) confirmed the points put forth earlier.

Combined sublimation of UF_6 and PuF_6 occurs in fluorination with nondilute fluorine and at higher temperatures (500-550°C). The combined sublimation of uranium hexafluoride and plutonium hexafluoride

may be resorted to as an independent process for obtaining mixtures of a certain composition, to be followed by the conversion of those mixtures into oxides suitable for the fabrication of fuel elements. The uranium and plutonium have to be separated in the vapor phase, however, in order to obtain pure uranium and plutonium compounds.

The thermal method of separating uranium and plutonium is based on the thermal instability of plutonium hexafluoride at temperatures above 150°C. The equilibrium constant of the thermal dissociation of plutonium hexafluoride at the temperature 600°K is 300, whereas the same constant for uranium hexafluoride has value of the order of 10^{-22}, i.e., the probability of UF_6 decaying under such conditions is virtually negligible.

Investigations of the kinetics of thermal decomposition $PuF_6 \rightarrow PuF_4 + F_2$ demonstrated that the rate of decay at temperatures 300-350°C is fully acceptable for technological purposes.

In the design of process equipment for dealing with processes involving the reprocessing, shipment, and condensation of plutonium hexafluoride, we have to take into account the proclivity of plutonium hexafluoride to decompose when bombarded by its intrinsic α-radiation, and to then interact with the walls of the process equipment, which will have the effect of bringing about plutonium losses in the train of equipment, piping, and valves.

The problem of the total plutonium balance in the fluoride fuel recovery scheme cannot be considered as solved as of this writing. Still to be solved are the problems of how to remove the decomposition products and how to cope with the corrosive behavior of the plutonium hexafluoride.

Conversion of Uranium and Plutonium Fluorides to Oxides

Since the end products of the fuel recovery cycle are uranium oxides and plutonium oxides, the need arises to convert the uranium fluorides and plutonium fluorides to the respective oxides. The simplest and most promising method for obtaining oxides of uranium and plutonium is the pyrohydrolytic method based on the interaction between the fluorides of uranium and plutonium and superheated steam. When fluorides of higher valency are to be converted, the presence of hydrogen in the reaction sphere is mandatory.

Depending on the sequence in which the operations in the fluoride method of fuel recovery are carried out, the resulting mixture of uranium and plutonium hexafluorides of specified ratio (say, U : Pu = 4 : 1) can be directed to the pyrohydrolysis operation to obtain a UO_2–PuO_2 mixture of specified composition.

This is the most logical and the simplest approach. The method in question has been developed intensively both in the USSR and in other countries, that being the case. The principal difficulty to be faced in the realization of this process is how to obtain a uniform homogeneous solid solution of the oxides of uranium and plutonium.

Process Design and Equipment Layout Problems in the Fluoride Method

At the present time, close attention is being given, in the USSR, to the development of process equipment designs for the fluoride method, and to searches for the most efficient equipment configuration in "hot" caves. Along with the devising of the fluoride method has come the need to work out design solutions for remote-controlled process equipment and instrumentation which are novel in principle. For example, remote-adjustable joints were developed for assembly and dismantling of equipment in "hot" caves. Joints utilizing low-melting materials were tried out in one of the variants to remote-adjustable joints between equipment parts.

Remote control of processes, valve operation, moving parts, and other components of equipment in the flowsheet is found to be a more complicated problem than the water technology.

The development of the fluoride method of spent fuel recovery necessitated the application of methods for shipping solid radioactive materials other than by water routes. In particular, the pneumatic method of transporting powdery products, which lends itself to realization in "hot" cave work, has found widespread application.

Tests of the fluoride method have pointed out the need to develop special valving equipment, process monitoring equipment and instrumentation, and special components for remote-control valving and instrumentation.

Several possible variants of process equipment and instrumentation configurations for the fluoride method of spent fuel recovery have been examined, with the object of settling upon an optimum variant which will combine reliable control of the technological process with observance of public health rules and regulations regarding work rooms frequented by personnel.

Several structural materials for the fabrication of equipment to be used in the fluoride technology have been tested. Nickel was found to be the most acceptable such material.

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EFFICIENT UTILIZATION OF FUEL IN FAST POWER REACTORS

A. I. Leipunskii, V. V. Orlov,
V. B. Lytkin, M. F. Troyanov,
and L. N. Yurova

It is well known that the supply of uranium on the earth is sufficient to satisfy the energy demands of mankind for the foreseeable future through the use of fission reactors. However, according to current ideas only a negligible fraction of this supply can be called economically suitable for use in thermal reactors. Light-water reactors (LWR), which now serve as the basis for the development of nuclear power in most countries, consume 0.6 kg of U^{235} per MW (el.) per year. At present-day prices for natural uranium $C_U \approx \$20/kg$ this leads to 0.04 cent/kWh as the uranium ore component of the cost of electric energy. This is approximately an order of magnitude smaller than the cost of fossil fuels and makes for inexpensive nuclear power.

Going over to uranium at \$50-\$100/kg means a significant increase in the expenditure for atomic electric energy.

Thus nuclear power, introduced to solve the fuel-power problem, now finds itself facing the same kind of problem.

There are two basic approaches to the solution of the nuclear fuel problem. Some authorities propose to invest capital in the search for new deposits, the development of inexpensive methods of extracting uranium from low-grade ore, the improvement of thermal reactors, and the development of the thorium fuel cycle. Others, including the present authors, see the cardinal solution of the fuel problem in fast neutron reactor-multipliers.

In the future, when these reactors have attained a certain breeding rate, they will be capable of making huge reserves of uranium and thorium available for the power industry, but even in the next few years the introduction of fast reactors will permit the development of a large-scale power industry, within the limits of the known cheap uranium resources.

From research and development work performed in several countries during the last two decades it appears likely that the problems of fast sodium-cooled reactors, including their large-scale use in power plants, will be solved within the next few years. Fast reactors now form the basis of promising programs for the development of nuclear power.

The introduction of fast reactors with breeding ratios BR > 1 still does not solve the nuclear fuel problem. To do this it is necessary that the BR be enough larger than unity to ensure secondary fuel to reload existing reactors and to start up new ones. In other words, the natural rate of growth of a system of fast reactors ω_0 must be at least equal to the rate of growth of nuclear power ω. In order that fast reactors can replace thermal reactors, so long as the supply of cheap uranium has not been used up, their natural growth must be enough larger:

$$\omega_0 > \omega. \tag{1}$$

The present article is devoted to a determination of the requirements for fast reactors arising from the problem of conserving our uranium resources, and to an investigation of their possibilities from this point of view. We limit ourselves to the discussion of sodium-cooled fast reactors with ceramic fuel since

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they are the most fully developed. We consider only those measures for improving the utilization of fuel which are being studied seriously at the present time. This practical attitude is justified by the fact that these improvements are turning out to be adequate.

The evaluation and optimization of reactors from the point of view of their consumption of natural uranium in comparison with the known uranium reserves is sometimes contrasted with the economic approach. Economic criteria permit taking complete account of the various expenditures for the production of electrical energy and in the final analysis determine the choice of reactors. However, these considerations at present uranium prices do not adequately reflect the necessity of conserving uranium resources since they do not take into account one of the most important economic advantages of fast reactors – the possibility of increasing power without significantly broadening the fuel base during the next decades. This justifies our concern with natural indexes of the consumption of uranium together with economic criteria.

Since for optimization it is desirable, and in a rigorous formulation necessary, to have a single criterion, we shall try henceforth to combine the natural and economic criteria, starting from the assumption of limited reserves of cheap uranium.

The development of nuclear power can be arbitrarily divided into two stages. In the first stage nuclear power develops rapidly and occupies a modest place in the power system. In the second stage it has become one of the main power sources and its rate of growth decreases to that of the total power demand. We consider first the requirements for fast reactors to become the predominant power sources of the distant future.

Nuclear Power Prospects

Allowing for the levels of future power demands it must be required that fast reactors be able to supply their own fuel, i.e., that their natural rate of growth be equal to the rate of growth of power:

$$\omega_0 = \omega.$$

In this case nuclear power requires uranium only as raw material for fast-reactor blankets. For a long time, as long as enriched fuel is used, this will not require additional consumption of uranium since the dumps of enrichment plants will be quite sufficient. The problem of natural uranium, and perhaps also of thorium, as a raw material will arise much later and is completely solvable since it is economically admissible to use very expensive uranium for this purpose. The magnitude of ω_0 depends on the characteristics of the reactors themselves, the fuel cycle, and the operating conditions in the power system, i.e., the load factor φ. In order to separate out this dependence we introduce as a characteristic of the reactor itself the quantity $T_2^{0.8}$, the doubling time for $\varphi = 0.8$. Then

$$\omega_0 = \frac{\ln 2}{T_2^{0.8}} \cdot \frac{\varphi(1+0{,}87_f/T_c)}{0.8(1+\varphi T_f/T_c)} \tag{2}$$

and the required doubling time as a function of ω and φ is

$$T_2^{0.8} = \frac{\varphi(1+0.8T_f/T_c)\ln 2}{0.8(1+\varphi T_f/T_c)\omega}. \tag{3}$$

Here T_f is the external fuel cycle time and T_c is the core lifetime for $\varphi = 1$.

When the nuclear power industry becomes large-scale it will have to take on, at least in part, the function of controlling power in power systems. Although the relatively low fuel costs and high capital expenditures for atomic power plants will stimulate attempts to prevent φ from decreasing, we shall assume that the load factor lies in the 0.6 to 0.8 range.

We list below the required values of $T_2^{0.8}$ as a function of the rate of growth of power for values of φ in the range indicated:

ω, 1/yr	$T_2^{0.8}$, yr
0,10	6—7
0,08	7—9
0,06	9—11

Thus even present-day fast reactors with oxide fuel, which can have a doubling time $T_2^{0.8} = 9$ yr for an external fuel cycle time $T_f = 0.5$ yr, ensure a rate of growth of power of ~6% per year.

TABLE 1. Characteristics of Prospective High-Power Fast Reactors

Parameters	FOR-2	FSR	FOB-1	FOB-2	FCB
Thermal power, MW	3750	3750	3750	3750	3750
Electric power, MW	1500	1500	1500	1500	1500
Volumetric heat release rate, kW/liter	650	760	550	540	650
Method of flattening heat release	Two enrichments	From data in [6]	Two enrichments	Composition	Two enrichments
Coefficient of nonuniformity:					
radial	1,27	1,18	1,37	1,26	1,3
axial	1,26	1,27	1,24	1,25	1,25
Core loading, tons	3,45	2,50	2,93	2,85	2,74
Specific core loading, tons/million kW (el.)	2,3	1,65	1,95	1,90	1,83
Fuel element diameter, mm	5,8	5,8	6,8	5,8–6,25	7,0
Breeding ratio	0,9	1,1	1,4	1,38	1,55
Doubling time for a load factor of 0.8:					
$T_f = 1$ yr	—	—	9	9,5	7,3
$T_f = 0.5$ yr	—	—	7,5	7,5	6

Higher rates of growth of power, characteristic of our country at the present time ($\omega \approx 8\%$ per year), require improvement of oxide reactors or a shift to new fuel forms such as monocarbide, which decreases $T_2^{0.8}$ to 6-8 yr. Therefore there is no doubt of the possibility of using fast reactors of the type now being developed to ensure the prospects of the development of nuclear power.

The question of higher rates of increase of nuclear power in the next decades is more real. In order to be able to change over to the construction of fast breeders only and to cease building thermal reactors in the foreseeable future the natural rate of increase of fast breeders ω_0 must be somewhat larger than ω. An analysis of the development of a system of thermal converters – fast breeders shows that the time for such a changeover is

$$t_0 \approx \frac{1}{r_T/g_f - (\omega_0 - \omega)} \ln \frac{r_T/g_f}{\omega_0 - \omega}. \quad (4)$$

Here r_T is the specific production of plutonium in the thermal converter and g_f is the specific loading of plutonium in the fuel cycle of the fast breeder.

For $r_T \approx 0.2$ kg per MW (el.) per year and $g_f \approx 4$ kg per MW (el.) nuclear power can be developed in such an arrangement (i.e., the construction of thermal reactors can be terminated) in a foreseeable time ($t_0 \approx 30$ yr) if $\omega_0 - \omega \approx 0.2$, i.e., $\omega_0 \approx 10\%$ per year, so that the required doubling times are 6-7 yr. The formula for the doubling time in its most complete form was obtained in [1]:

$$\bar{g}_c\left(1+\varphi\frac{T_f}{T_c}\right)+\bar{g}_{ab}\left(1+\varphi\frac{2T_f}{T_c}\right)+\rightarrow \quad T_2=\frac{\rightarrow +\bar{g}_{rb}\left(1+\varphi\frac{T_f}{T_{rb}}\right)}{\varphi[\gamma(KB-1)-Y(1-\varepsilon)]}\ln 2. \quad (3')$$

Here $\bar{g}_c$, $\bar{g}_{ab}$, and $\bar{g}_{rb}$ are respectively the average amounts of plutonium in the core, axial blanket, and radial blanket per unit power; y is the amount of plutonium burned up per year unit power; Y is the total amount of plutonium which can be unloaded from the reactor per year per unit power; ε is the coefficient of recovery from conversion, taking account of losses; T_{rb} is the average delay time of plutonium in the radial blanket; and BR is the total breeding ratio of the reactor.

The main ways of decreasing T_2 are:

1) a decrease in the amount of plutonium in the fuel cycle by shortening the external cycle time;

2) a decrease in the amount of fuel in the cycle as a result of increasing the specific power of the fuel by flattening the heat release distribution;

3) an increase in breeding by going to new forms of fuel and improving fuel elements and assemblies;

4) an increase in the fuel burnup;

Calculations showed that those improvements in fast reactors can reduce the consumption of natural uranium for the development of nuclear power to the level of known resources of cheap uranium.

Further saving of uranium, if required, can be made by improving thermal reactors or by replacing some of the LWR with fast uranium converters, particularly in the early stages.

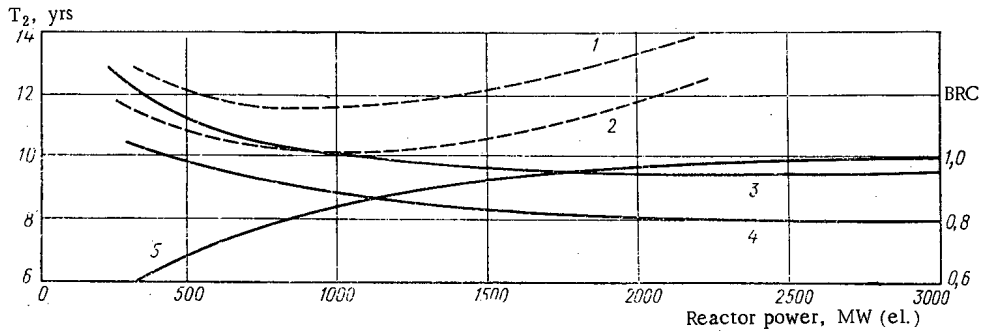

Fig. 1. Doubling time T_2 and internal breeding ratio BRC as functions of electric power of reactor for external fuel cycle times T_f of 1 and 0.5 yr (fuel UO_2 $-PuO_2$, maximum 10^5 MW days/metric ton): 1) doubling time T_2 without flattening of heat release distribution, T_f = 1 yr; 2) T_2 without flattening of heat release distribution, T_f = 0.5 yr; 3) T_2 with two-zone flattening of heat release distribution, T_f = 1 yr; 4) T_2 with two-zone flattening of heat release distribution, T_f = 0.5 yr; 5) internal breeding ratio (BRC).

In the next section we analyze methods of flattening the radial heat release distribution in the core and stabilizing it in time to increase the specific power in high-power reactors and decrease the consumption of fuel.

High-Power Fast Reactors

We can expect the power levels of individual fast reactors in atomic power plants to rise to 1000-2000 MW (el.) in the near future. This lowers the specific capital investments and improves the economics of the fuel cycle [2].

The principal characteristics of the physics of high-power fast reactors depend upon the following factors:

1) the increase in the internal breeding ratio reactivity and the fact that it is close to unity. This simplifies the problem of compensating the change in reactivity with burnup and makes possible long continuous reactor operation without refueling (up to ~1 yr or more);

2) the necessity of flattening the profile of the heat release distribution in the reactor core. This ensures a decrease in the relative amount of fuel in the cycle and constancy of the total breeding ratio with increase in reactor power.

At the present time the most widely used method of flattening the heat release distribution involves making zones of different enrichment in the reactor core. The dependence of the physical characteristics of reactors on their power has been studied by S. T. Leskin and A. I. Novozhilov.

The results of these investigations are shown in Fig. 1. It is clear from the figure that the physical characteristics of the reactor improve as the power increases.

For a reactor of the BN-600 type having oxide fuel and with the heat release distribution flattened by zones of different enrichment the doubling times are 9 and 11 yr for external fuel cycle times of 0.5 and 1 yr. Nuclear power development interests require these times to be 6-8 yr.

The study of the potentialities of fast reactors requires taking account of certain technical progress in reactor construction such as improvements in the construction of fuel assemblies and elements, improvements in technology, and the development of new fuel composition [2].

On the other hand, when the heat release distribution is flattened by making zones of different enrichments, the profile of the heat release distribution changes appreciably when the reactor is operated at high power for long times without refueling (0.5 yr and more). During the operation of such a reactor there is a relative increase in the heat release at the center of the core because of the spatial redistribution of the concentrations of fissionable isotopes and of the neutron flux, which in the last analysis determines the duration of continuous reactor operation. Problems of the time instability of the heat release profile have been studied with the assistance of S. B. Boborv, V. M. Murogov, A. I. Novozhilov, L. V.

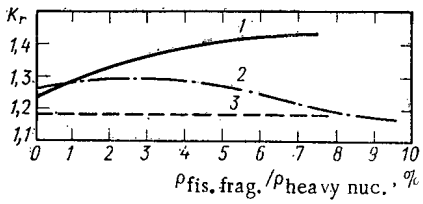

Fig. 2. Radial coefficient of nonuniformity of heat release K_r as a function of the time of continuous reactor operation: 1) flattening by two enrichments; 2) flattening by composition; 3) reactor with stable heat release distribution.

Tochenyi, A. N. Shmelev, V.G. Ilyunin, and I. D. Rakitin, et al. [3-5]. S. M. Zaritskii, V. M. Murogov, and A. N. Shmelev developed a method for flattening the profile of the heat release distribution in fast reactors which ensures its time stability [6] by making subzones with different fissionable and breeding isotopes.

Using this method of flattening the heat release distribution E. I. Grishanin, V. M. Murogov, V. V. Orlov, and L. V. Tochenyi proposed a fast power reactor with a time-stable profile of the heat release distribution (FSR – fast stable reactor).

Investigations performed with the assistance of V. G. Ilyunin, A. M. Kuz'min, V. M. Murogov, A. N. Shmelev, and Yu. V. Silaev showed that the heat release distribution in such a reactor is flattest and changes least with reactor operation (Fig. 2). This ensures the possibility of prolonged continuous reactor operation without refueling, and an appreciable increase in the power density in the fuel in the fuel cycle. Figure 2 illustrates the change in the radial coefficient of nonuniformity K_r with burnup for various flattening procedures in a reactor operating without refueling. The behavior of the heat release distribution in a plutonium reactor flattened by enrichment is significantly different from that in a stable reactor.

The method of flattening by composition – varying the fuel element pitch or diameter in different zones – has other advantages in high-power fast reactors.

As a result of studies by A. I. Novozhilov, S. B. Bobrov, V. G. Ilyunin, A. M. Kuz'min, V. N. Murogov, Yu. V. Silaev, and A. N. Shmelev it has been established that the stability of the heat release distribution in flattening by composition, achieved by the equality of the zonal breeding ratios, ensures a rather high breeding rate and power density of the fuel in the cycle (Fig. 2). This method of flattening leads to more negative values of the sodium and Doppler coefficients of reactivity which simplifies certain safety problems.

Work performed with the assistance of S. B. Bobrov and A. I. Novozhilov showed that in flattening by enrichment an optimum distribution of plutonium fuel with a different content of the higher plutonium isotopes (Pu^{240}, Pu^{241}, Pu^{242}) considerably improves the time behavior of the heat release distribution and increases the power density and the breeding rate.

A proper and more all-round study of various flattening methods is impossible without performing a complex fast reactor core calculation involving thermal, hydraulic, strength, and neutron physics calculations in combination. These problems were solved successfully by A. M. Kuz'min, A. A. Kashutin, Yu. V. Silaev, L. V. Tochenyi, V. V. Khromov, and V. A. Apse who developed a complex program for fast reactor optimization studies [7-9]. The ROKBAR optimization program permits the complex calculation of a sodium-cooled fast reactor core and determines compositions and reactor parameters which are optimum by one criterion or another: core dimensions, fuel element lattice pitch, fuel element cladding diameter and thickness, coolant velocity, volume fraction of assembly walls, height of gas cavity, fuel enrichment.

Constraints are imposed on certain parameters functionally dependent on controls, for example, the temperature of the fuel and cladding of the most highly stressed fuel element, taking account of hot spots, stresses in the fuel element cladding and assembly walls, the average heating of the coolant taking account of orificing.

Physical calculations are performed in the ROKBAR program by an effective few-group method with arbitrary space and energy intervals. An optimum variant of a reactor is sought by an iteration method. At each step of the iteration the initial problem is linearized by using the theory of small perturbations, and the linear problem is solved by the method of successive reduction of residuals [10].

Table 1 lists the results of an optimization study of large fast reactors by S. B. Bobrov, V. G. Ilyunin, A. M. Kuz'min, V. M. Murogov, A. I. Novozhilov, Yu. V. Silaev, and A. N. Shmelev which shows the characteristics of high-power (1500 MW (el.)) fast reactors with various forms of flattening, taking account of anticipated progress in reactor engineering: FOR-2 (fast oxide reactor), an advanced uranium oxide reactor flattened by enrichment; FSR (fast stable reactor), an advanced fast reactor with a stable heat release

TABLE 2. Reference Technological and Structural Data for Optimization Calculation

Parameter	Dioxide	Monocarbide
Maximum fuel temperature, °C	2450	1600
Maximum cladding temperature, °C	725	725
Fuel density, g/cm³	8,7	11
Cladding thickness, mm	0,4	0,3
Cladding material	Stainless steel	Stainless steel
Maximum burnup, MW days/metric ton	100 000	100 000
Type of fuel element	Sealed	Vented
Maximum strain of cladding	0,002	0,002
Pressure drop in reactor, abs. atm	12	12
Thermal conductivity of fuel, kcal/m·h·°C	2,2	16

distribution; FOB-1 (fast oxide breeder), an advanced plutonium oxide reactor flattened by enrichment; FOB-2, an advanced plutonium oxide reactor flattened by composition; FCB (fast carbide breeder), an advanced plutonium carbide reactor flattened by enrichment.

The initial technological and structural parameters, obtained mainly by optimization, are listed in Table 2.

Some of the reference data taken here for the optimization calculation assume a certain amount of technical progress, such as an appreciable improvement in the hydraulic resistance of fuel assemblies and the construction of vented fuel elements. The moderately promising value of 10^5 MW days/metric ton is used for maximum burnup. In this connection it should be noted that the recent discovery of the swelling of irradiated steel can require further technical measures for achieving the assumed burnup.

Consumption of Natural Uranium

We assume that the development of nuclear power is described by the expression

$$N(t) = \frac{A}{\omega}(e^{\omega t} - 1) + N(0), \tag{5}$$

where N(t) is the power level at time t; A is a constant equal to 2.38 in the present case; ω is the specified asymptotic rate of increase of power equal to 0.08 in the present case; and N(0) is the initial power level equal to 2 in the present case. The time interval considered is 30 yr. Calculations of the consumption of uranium have been made with the help of L. S. Anufrienko.

Figure 3 shows the variations in the consumption of natural uranium over the next 30 yr obtained by using combinations of uranium and plutonium reactors with different characteristics.

Curve 1 gives the consumption of uranium for uranium and plutonium reactors (FOB + FOR) with "contemporary" characteristics; curve 2 refers to fast reactors with stable heat release distributions (FSR); curve 3 shows how markedly the consumption of uranium can be reduced by using a monocarbide breeder with promising characteristics (FCB). In all three variations the load factor is taken as 0.8 and the concentration of U^{235} in depleted uranium as $x_0 = 0.25\%$. The external cycle time is constant and equal to 1 yr. If the external fuel cycle were shortened to 0.5 yr, which is a technical possibility, the consumption of uranium could be decreased by another 25-35%.

The variations considered above are largely of theoretical interest for the analyses of fast reactor possibilities. Actually nuclear power development begins with thermal reactors since they have been worked on the most and form the basis of the nuclear power industry in all countries. It is well known that the development of nuclear power using thermal reactors with circulating plutonium would require appreciably more natural uranium than the development of power with fast plutonium reactors, since plutonium is used inefficiently in thermal reactors. This difference must increase with increasing length of the time interval considered. This effect will show up strongly in the next 30 yr.

Figure 4 shows curves of the total consumption of natural uranium for three variations of the development of nuclear power according to Eq. (5). The numbers are shown for illustrative purposes only.

The thermal reactors taken are two variations of light-water reactors: contemporary LWR-1 and prospective LWR-2; their characteristics, given in Table 3, correspond approximately to data in [11]. It is assumed that for all atomic power plants the load factor is 0.8 for the first 10 yr and 0.7 for the next 20 yr.

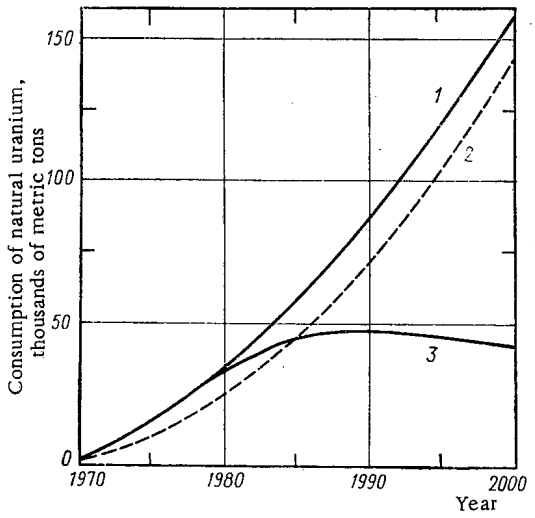

Fig. 3

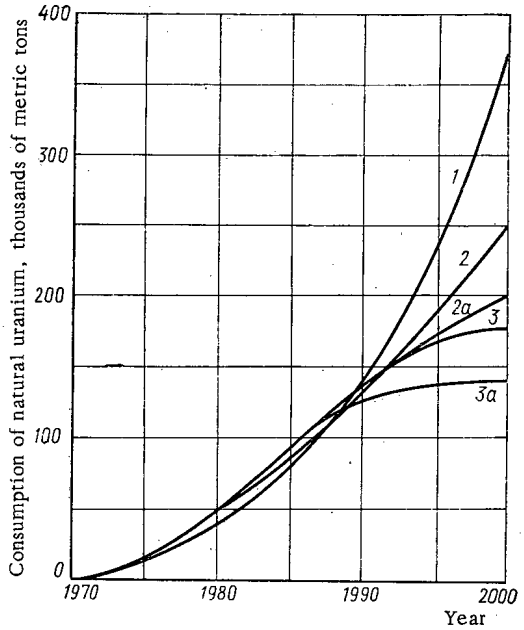

Fig. 4

Fig. 3. Total consumption of natural uranium as a function of fast reactor characteristics: 1) FOR + FOB; 2) FSR; 3) FOR + FCB.

Fig. 4. Total consumption of natural uranium for various combinations of reactors: 1) LWR-1 + LWR-2; 2) LWR-1 + LWR-2 + FOB + FCB; 2a) LWR-1 + LWR-2 + FOB + FCB (T_f = 0.5 yr); 3) LWR-1 + FOR + FOR-2 + FOB + FCB; 3a) LWR-1 + FOR + FOR-2 + FOB + FCB (T_f = 0.5 yr).

If there are fast reactors in the system the load factor after the first 10 yr is decreased only for atomic power plants with thermal reactors; fast reactors operate with a load factor $\varphi = 0.8$ at all times; depending on the number of thermal reactors φ for them is decreased by such an amount that the overall average load factor is ~0.7. Unless specifically stated, the external fuel cycle time is taken as 1 yr.

We consider the following variations (Fig. 4).

1. Light-water reactors LWR-1 for the first 10 yr plus light-water reactors LWR-2 for the next 20 yr; the plutonium produced is used in these same reactors; the effect of using plutonium is taken into account by decreasing the total consumption of uranium by 20%.

2. Light-water reactors LWR-1 for the first 10 yr plus light-water reactors LWR-2 for the next 20 yr together with fast plutonium reactors; fast plutonium reactors with oxide fuel (FOB) begin to be introduced after the first 10 yr, and 5 yr after that fast plutonium reactors with carbide fuel (FCB) instead.

2a. As in variation 2 but with an external fuel cycle time T_f = 0.5 yr.

3. Light-water reactors LWR-1 are built for the first 10 yr and then instead of them fast uranium reactors (FOR) are built together with fast breeder reactors (FOB) as needed; 15 yr after the beginning of the development of power, carbide breeders (FCB) are built instead of oxide breeders (FOB), and improved FOR-2 reactors as required instead of FOR.

3a. As in variation 3 but with the external fuel cycle time reduced from 1 to 0.5 yr after 1985.

Calculations show that there is a somewhat smaller total consumption of uranium in the variations involving thermal reactors than in the other variations only for approximately the first 17 yr, but toward the end of the second decade all the consumption curves intersect and after this the consumption of uranium becomes larger in the variations with only thermal reactors than in the other variations, and this difference increases rapidly with time.

Toward the end of the third decade the consumption of uranium in the variations with fast reactors is 33-62% less than in the "all thermal" variation 1.

The variations which consume the least uranium are the ones in which during the first 10 yr fast uranium reactors instead of thermal uranium reactors are constructed as needed to supplement fast breeders.

TABLE 3. Characteristics of Thermal Reactors

Parameter	LWR-1	LWR-2
Loading, metric tons of natural uranium/million kW (el.)	650	400
Consumption for makeup (φ = 0.8), metric tons of natural uranium/million kW(el.)year	160	120
Plutonium production (φ = 0.8), metric tons of fissionable Pu /million kW (el.)year	0,200	0,200

The use of fast uranium reactors ensures the smallest consumption of uranium not only for 30 yr but also during the subsequent period. We have in mind the consumption of uranium for makeup in atomic power plants with uranium reactors started up after the end of the 30 yr period considered. This consumption will be somewhat larger for variation 1 with only thermal reactors than for variations 3 and 3a where there are few thermal reactors, and besides after the year 2000 there is the possibility of using surplus plutonium from fast reactors for makeup.

As has already been noted, the maximum burnup values assumed here for prospective fast reactors are moderate and are already attainable with oxide fuel.

An increase in burnup thus remains as a possible reserve for a future increase in the efficiency of the fuel cycle. We do not consider here the possibility of a further improvement in characteristics by using metallic fuel since there are not sufficient experimental data on its permissible temperature, density, alloying, burnup, etc.

As regards the effect of the nuclear power development scale on our calculation of the natural uranium requirements, we can presume that, within certain limits, the final total figures will be proportional to the power level at the end of the period considered.

The effect of inaccuracies in nuclear data on calculations of fast reactor characteristics and the consumption of natural uranium remains to be discussed. The present calculations of physical characteristics of reactors were based on a 26-group set of constants BNAB [12] revised in 1969 to conform with the most recent experimental data on Pu^{239}, U^{238}, and U^{235} [13-15].

Correcting the cross sections increased the specific loading by about 6% and decreased the breeding ratio for the types of fast reactors considered by 0.05. The reactor characteristics listed in Table 1 have taken these changes into account.

It was estimated in [16] that such characteristics of a large fast reactor as the breeding ratio can be calculated to ± 0.1, and the critical mass to $\pm 8\%$. This is explained by the uncertainty of the nuclear data. If reactor characteristics are calculated by using pessimistic estimates, i.e., a decrease in the breeding ratio of 0.1 and an increase in core loading by 8%, the 30 yr consumption of uranium in a system with thermal and fast reactors is increased by no more than 8-15% over the figures given in Fig. 4.

Economic Criteria of the Efficiency of Utilization of Nuclear Fuel

As was noted above, the economic criteria should take account of the fact that the supply of cheap natural uranium is limited and that its price will increase with time. Reactor characteristics are affected by the rate of consumption of uranium and variations in its price, and these react on the economic indexes of reactors.

In determining variations in uranium prices we shall not take into account the inevitable "normal" technological progress leading to a decrease in expenses since in optimizations and comparisons it is not the absolute but the more stable relative magnitudes which are important.

If G(t)dt is the consumption of natural uranium in time dt and g(C) is the uranium supply, the expenditure for which does not exceed \$C/kg, C(t) can be obtained from the expression

$$g[C(t)] = \int_0^t G(t)\,dt. \tag{6}$$

In estimating the economic efficiency we start from the present value of the total expenditures for natural uranium:

$$Z = \int_0^\infty dt e^{-pt} G(t) C(t), \tag{7}$$

where p is the discount rate.

By varying Eqs. (6) and (7) it is found that for a change $\delta G(t)$ in the consumption of uranium the change in total expenditure is

$$\delta Z = \int_0^\infty dt e^{-pt} \delta G(t) \tilde{C}(t), \tag{8}$$

where

$$\tilde{C}(t) = C(t) + \int_t^\infty dt' e^{-p(t'-t)} \frac{dC}{dt}. \tag{9}$$

The quantity

$$\tilde{C}(t) = e^{pt} \frac{\delta Z}{\delta G(t)} \tag{10}$$

should be taken as the price of uranium under conditions of limited resources. It takes account of the expenditure at time t for a kilogram of uranium and for the increase in expenditure in the future for the consumption of a supplementary kilogram of uranium. Similarly the price of plutonium is

$$\tilde{C}_{Pu}(t) = e^{pt} \frac{\delta Z}{\delta G_{Pu}(t)} = \int_t^\infty dt' e^{-p(t'-t)} C(t') \frac{\delta G_U}{\delta G_{Pu}} \approx \alpha\gamma C(t), \tag{11}$$

where α is the relative efficiency of Pu^{239} and U^{235} in the reactor; $\gamma = (1 - x_0/x)/(0.0071 - x_0) \approx 200$; x_0 is the concentration of U^{235} in depleted uranium; and x is the concentration of U^{235} in enriched uranium. Here δZ represents the saving in expenditures for uranium by introducing an additional kilogram of plutonium at time t. The quantities δZ and $\tilde{C}_{Pu}$, of course, depend on how the plutonium is used, e.g., in thermal or in fast reactors, and therefore can be uniquely determined only for a specific system. We emphasize that $\tilde{C}_{Pu}(t)$ determines only one component of the price of plutonium, related to the consumption of natural uranium.

The discounted expenditure for natural uranium for a reactor of the i-th type is

$$Z = e^{pt} \frac{\delta Z}{\delta N_i} = \gamma [g_i \tilde{C}(t) + q_i \int_t^\infty dt' e^{-p(t'-t)} \tilde{C}(t') - r_i \int_t^\infty dt' e^{-p(t'-t)} C_{Pu}(t')], \tag{12}$$

where g_i (kg/MW), q_i (kg/MW/yr), and r_i (kg/MW/yr) are the specific consumptions of fuel (U^{235} or Pu^{239}) for loading, makeup of reactor, and production of surplus plutonium respectively, and δN_i is the increase in power. With limited resources the cost of generating electric power increases with time simultaneously with the price of uranium. A similar analysis shows that the uranium component of the cost of 1 kWh has the form

$$S = \left(p\tilde{C} - \frac{d\tilde{C}}{dt}\right) \gamma g + \gamma q \tilde{C} - r\tilde{C}_{Pu}, \tag{13}$$

which when integrated leads to the total discounted expenditure (12).

Expenditures calculated in this way take into account the change in the price of uranium as a consequence of limited resources, and in this sense combine economic and natural criteria. Taking account of the increase in price leads to an increase in the absolute magnitude of the expenditures and to a relative decrease in the role of the initial loadings.

The formulas obtained indicate an increase in the economy of thermal reactors with the start-up of fast reactors because of the increase in the price of plutonium and also because of the feasibility of shifting their construction and operating conditions toward an optimum production of plutonium.

Let us make a crude quantitative estimate of the changes in the economic indexes of reactors as a result of taking account of limited uranium resources.

In the version of the development of nuclear power on a global scale with only light-water reactors, the expenditures for natural uranium increase in 30 yr from C = 20 to C = \$60/kg, so that on the average $dC/dt \approx$ \$1.3/kg·yr.

At present-day prices the discounted expenditure for uranium in LWR reactors is ~$30/kW for p = 6%/yr, and by Eqs. (7) and (4) it is ~$80/kW. For the development of power with LWR and fast reactors for p = 10%/yr these figures are 20 and $40/kW, respectively.

Expenditures for fuel for fast reactors turn out to be smaller (within the limits ±$10/kW) and more weakly dependent on the price of uranium. Expenditures calculated by using Eqs. (9), (12), and (13), together with expenditures for enrichment, construction, chemical reprocessing of fuel elements, etc., can serve as criteria of economic efficiency in the optimization of reactors under conditions of limited uranium resources.

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PROPERTIES OF CARBIDE, NITRIDE, PHOSPHIDE, AND OTHER FUEL COMPOSITIONS AND THEIR BEHAVIOR UNDER IRRADIATION

T. S. Men'shikova, F. G. Reshetnikov, V. S. Mukhin, G. A. Rymashevskii, I. G. Lebedev, and A. L. Epshtein

A great deal of attention is now being paid to the study of carbide fuel for nuclear reactors; this is superior to oxide fuel in respect of a number of properties (higher thermal conductivity, greater density with respect to the fissile element, etc.). The use of a carbide fuel may considerably improve such an important characteristic of fast reactors as the doubling time, thus having a favorable effect on the economics of the fuel cycle. From this point of view, uranium and plutonium phosphides and nitrides are also interesting compositions.

Experience in the use of the BR-5 reactor, incorporating a carbide zone, in the USSR has confirmed the very real possibility of using carbide fuel in fast reactors. Other recent investigations [1] have shown, however, that carbide fuel also has a number of shortcomings: at high temperatures (1500°C) it swells more than oxide fuel and carburizes the can. Thus in order to realize the advantages of carbide fuel it is essential to develop a corresponding construction of the fuel element. In order to reduce the temperature of the core, for example, it is desirable to use an underlayer with a high thermal conductivity such as sodium in the fuel-can gap. However, the use of sodium worsens compatibility with the can, since sodium carries excess carbon and carburizes the can surface. There are nevertheless certain ways of eliminating or at least greatly reducing the carbiding of the can. These include: 1) producing carbides of very nearly stoichiometric composition; 2) alloying the carbides with elements which easily form compounds with carbon; 3) depositing protective coatings on the core, or protecting the can from carbiding in some other manner. Many investigations have been devoted to an examination of the relative merits of these methods under irradiation.

A fair number of investigations [1-8] have also been concerned with the physicomechanical, radiation, thermodynamic, and other properties of refractory uranium and plutonium compounds. However, further work still has to be done on the optimum methods of preparing refractory uranium and plutonium compounds and studying their properties, as well as problems of swelling, compatibility, and gas evolution under conditions of reactor irradiation. The present contribution is devoted to problems of this kind.

Production of Refractory Uranium and Plutonium Compounds

Refractory (oxygen-free) compounds of uranium and plutonium may be produced in various ways. Carbides of these metals, in particular, are mainly obtained by the reaction of the oxides with carbon. A second possible method is the gas carbiding of the corresponding metals. Despite the fact that the metals are more expensive than the oxides, the latter method has a number of advantages. The low temperatures of the processes, the simplicity of the apparatus, and the possibility of mechanizing and automating the production of the fuel-element cores make this method attractive and worth further consideration. We therefore considered it desirable to make a further study of various methods of producing uranium monocarbide and mononitride by carbiding or nitriding fine uranium powder or its hydride with propane or nitrogen, respectively.

State Committee for the Use of Atomic Energy of the USSR. Translated from Atomnaya Énergiya, Vol. 31, No. 4, pp. 393-402, October, 1971.

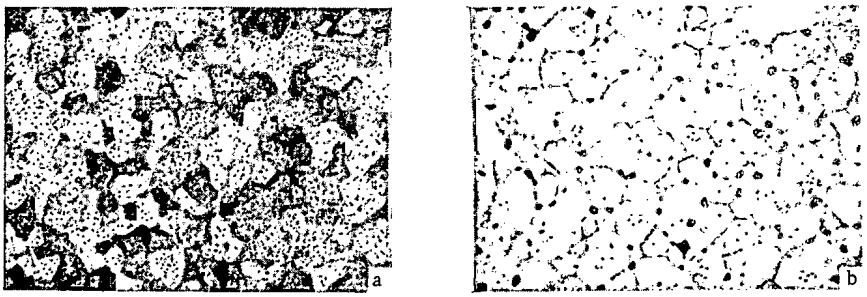

Fig. 1. Microstructure of moldings made by gas carbiding: a) Uranium carbide; b) uranium nitride (×200).

Investigations showed that, by varying the temperature and velocity of carbiding, uranium carbides of any specified composition might be obtained, from the hypostoichiometric monocarbide to the dicarbide. This method is capable of yielding stoichiometric uranium monocarbide on the large scale in the form of a powder with a 2-6 μ particle size and a lattice constant of a = 4.961 Å. The oxygen content is no greater than 0.05%.

Uranium mononitride was obtained by the interaction of fine uranium or uranium hydride powders with nitrogen at 450-700°C, with subsequent decomposition of the sesquinitride so formed in vacuo at 900-950°C so as to obtain the mononitride. The resultant mononitride contained 5.4-5.5% nitrogen; the lattice constant was 4.889 Å; the particle size averaged 15-30 μ. In making cores from the mononitride, sintering was carried out in a nitrogen atmosphere at 1800-2000°C. Uranium sesquinitride may be produced as a continuous process.

Plutonium monocarbide was prepared by the hydrogenation of metallic plutonium and its subsequent carbiding with carbon at 400-800°C. A solid solution of uranium monocarbide in plutonium monocarbide was obtained by mixing the original powders and then sintering at 1200-1600°C. The high surface activity of the resultant powders enabled us to simplify and automate the preparation of fuel-element cores in the form of moldings of various shapes with a high and controllable density (93-96% of theoretical) using an automatic molding system or multiple-position molds. The cores were prepared by mixing the original powders with 1% of a 1.5% solution of paraffin or oleic acid in ligroin and pressing at 1.5-2 tons/cm^2, followed by vacuum sintering at 1600-1800°C for 3-4 h. The resultant moldings had a density of 93-95% of theoretical and a mean grain size of 30-50 μ, with equiaxed crystals. In preparing the cores for the experimental fuel elements BOR-60 and SM-2 the amount of oxygen in the monocarbides and mononitride cores was respectively 0.02-0.05 and 0.1%. The characteristic microstructures of the compounds are shown in Fig. 1a and b. The uranium and plutonium sulfides and phosphides are obtained by the same method, carbon being replaced by hydrogen sulfide and phosphine respectively.

In order to obtain uranium and plutonium nitrides and carbides and produce cores from these, a special apparatus was developed; this comprises eight metal chambers in series, with additional equipment (electric furnaces to produce the original compounds, presses, vacuum-compression electric furnaces for sintering, balances, control desks, etc.). The general appearance of this installation is shown in Fig. 2.

Properties of Uranium Carbide, Nitride, Sulfide, and Phosphide

Mechanical and Thermophysical Properties: In order to refine the scattered and sometimes contradictory published data, we studied the properties in question, i.e., the thermal conductivity λ, the electrical resistivity ρ, the specific heat C_p, and the vapor pressure P_U of uranium (in general up to 2000°C), together with the compressive strength σ_b, the hardness H_v, the elastic modulus E, and the linear expansion coefficient α of UC, UN, U–C–N, US, and UP.

The samples were prepared metalloceramically and their porosity equalled 5-10% (the values given for λ, ρ, E, C_p correspond to the nonporous state); the proportion of nonmetallic components was almost stoichiometric (the structure of UP and US contained 2-5 wt. % of oxide inclusions).

The strength and other properties were measured in vacuum; the hardness was determined from the impression of a sapphire indentor with a load of 2 kg; the elastic modulus was determined by a resonance

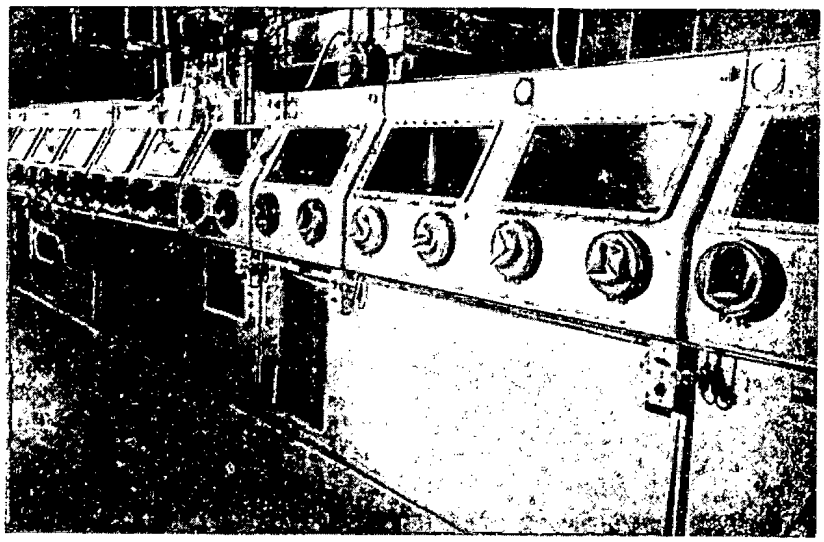

Fig. 2. General view of the apparatus for producing uranium carbides and nitrides.

method in samples with a ratio of $l/d \geq 10$. The vapor pressure was studied by an integrated version of the Knudsen method. The thermal conductivity at 20-1000°C was calculated from measurements of the specific heat and thermal diffusivity; for higher temperatures it was measured by the radial thermal-flow method (ρ being determined at the same time). The thermal expansion was studied in an induction quartz dilatometer.

The error in the determination of λ and P_U was no greater than 15%, in the elastic modulus 5%, and in the thermal expansion 3%.

Figures 3-5 show the temperature dependence of σ_b, E, H_v, $\overline{P}_U$, λ, ρ, C_p for uranium carbide, nitride, sulfide, and phosphide, and Fig. 6 shows the concentration dependence of the properties in the UC–UN system. Comparison of the temperature dependence of σ_b, H_v, E and also the deformation (Fig. 3) shows that the sulfide, and particularly the phosphide, are much more ductile than the carbides and nitrides. This appears both in the absence of any rise in strength with temperature, such as is characteristic of brittle phases, and also in the substantial deformation at T > 1000°C. Cracks around the hardness-measuring impressions failed to appear in the phosphide samples even at room temperature (in the other compounds the temperature at which the cracks vanished lay at 1000-1200°C). We notice the fairly low elastic modulus of uranium phosphide, which, combined with the "ductility" noted earlier, gives a heat resistance greater than that of uranium carbide and nitride. Our own determinations of σ_b, E, H_v are in fair agreement with published values [9-11].

The lowest uranium vapor pressure among all the compounds studied occurs in the case of the carbide and the highest in the case of the phosphide (Fig. 4). Expressions for the temperature dependence of the vapor pressure (mm Hg) take the form

$$\lg P_U = 7.56 - \frac{2.86 \cdot 10^4}{T}, \quad T = 2170 - 2470^\circ\,\mathrm{K}, \quad U\,(C_{0.99}N_{0.005}O_{0.005}),$$

$$g P_U = 8.43 - \frac{2.74 \cdot 10^4}{T}, \quad T = 1850 - 2300^\circ\,\mathrm{K}, \quad U\,(N_{0.98}O_{0.01}C_{0.01})$$

$$\lg P_U = 9.75 - \frac{2.84 \cdot 10^4}{T}, \quad T = 1940 - 2530^\circ\,\mathrm{K},\ \mathrm{US}.^*$$

$$\lg P_U = 11.34 - \frac{2.94 \cdot 10^4}{T}, \quad T = 1920 - 2230^\circ\,\mathrm{K},\ \mathrm{UP}.^*$$

The phosphide and sulfide are characterized by congruent evaporation; the uranium vapor pressure depends considerably on deviations from stoichiometrical composition. As regards uranium carbide, this relationship was discussed in an earlier paper [6] on the basis of experimental data [6, 12] and statistical

*The chemical composition of the sulfide and phosphide (wt. %) are: US – 11.8 S; 0.05 N; 0.1 C; 0.25 (UO_2 + UOS); 0.1 W; UP – 11.7 P; 0.03 C; 0.7 O.

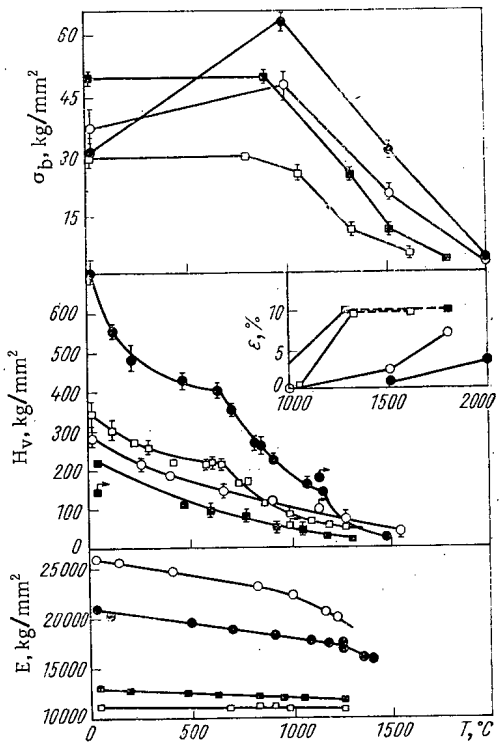

Fig. 3

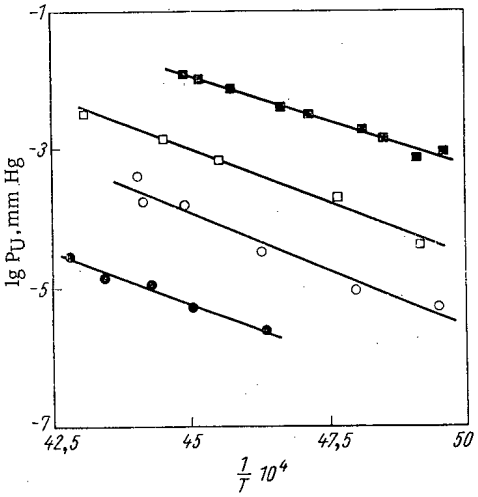

Fig. 4

Fig. 3. Effect of temperature on σ_b, H_v, E, and the deformation of uranium carbide (●), nitride (○), sulfide (□), and phosphide (■). The arrows indicate the temperature at which cracks vanish around the impressions during the hardness tests.

Fig. 4. Vapor pressure of uranium for uranium carbide, nitride, sulfide, and phosphide. Notation as in Fig. 3.

thermodynamics. The vapor pressure falls with increasing degree of completion of the nonmetallic lattice, allowing for the presence of all interstitial elements.

Our own data for uranium nitride are much lower (1-1.5 orders of magnitude) than the earlier results [6, 13] obtained on hypostoichiometric samples. The results of mass-spectrometric analysis of the uranium mono-compounds [12, 14-16] agree closely with our own. The electrical resistance, thermal conductivity, and specific heat rise with increasing temperature (Fig. 5); for uranium phosphide there is an anomalous change in ρ below 500°C and a nonmonotonic variation in λ and C_p in the neighborhood of 700°C. These anomalies also appeared earlier [3] for a case in which λ and C_p were measured up to 700°C. It is also a characteristic feature that in this range of temperatures there is a nonmonotonic variation in the coefficient of linear expansion, although the thermo-emf remains constant (~+50 μV/°C) at T = 20-1100°C. Judging from these data, uranium phosphide undergoes a transformation of the second kind at T ≈ 700°C; the nature of this transformation requires further attention.

The electrical resistivity of the phosphide, nitride, and sulfide are higher than that of the carbide; this is associated with the fact that the first three compounds undergo magnetic transformations at low temperatures [17]. Judging from the value of the Hall constants ($R_{UP} = +55 \cdot 10^{-4}$, $R_{UC} = +1.4 \cdot 10^{-4}$ cm^3/C), and also the thermo-emf coefficient, these compounds possess hole-type conductivity and may be regarded as semimetallic. The level of thermal conductivity in these materials in fairly high.

For the uranium carbonitrides, which form a continuous series of solid solutions, the properties vary quite smoothly with concentration (Fig. 6); the uranium vapor falls on passing from the nitride to the carbide [6]; the melting diagram [18] indicates a maximum close to the composition $UC_{0.3}N_{0.7}$ (T_m ~2900°C).

Compatibility of Uranium and Plutonium Compounds with the Can Material. A study of the compatability of refractory uranium and plutonium compounds with the can, both in direct contact and also through a helium and sodium interlayer, was carried out both before introducing the materials into the reactor and also under reactor conditions. Microstructural, x-ray structural, and chemical analyses were

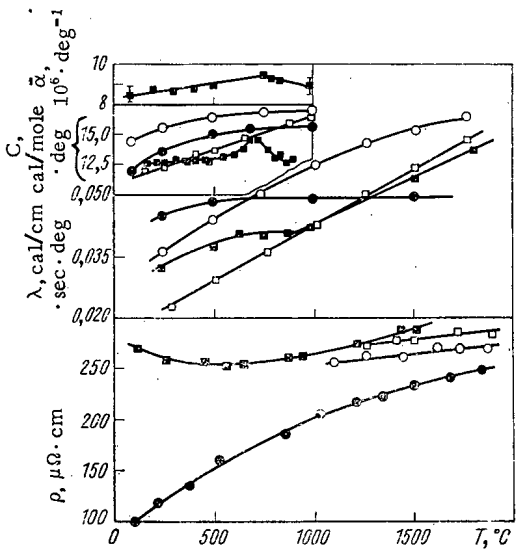

Fig. 5

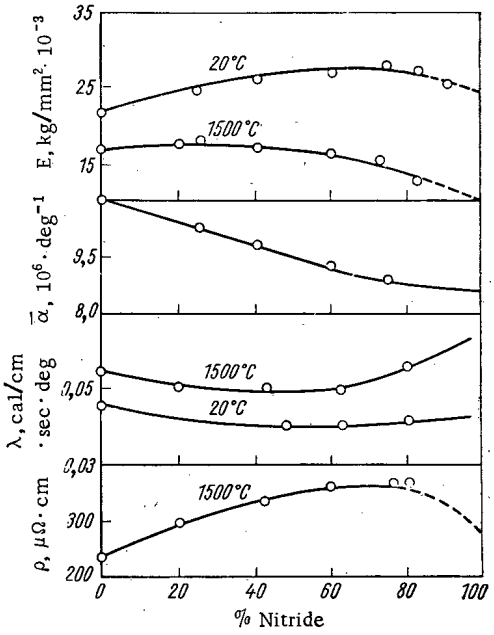

Fig. 6

Fig. 5. Temperature dependence of λ, ρ, C_p, $\bar{\alpha}$ for uranium carbide, nitride, sulfide, and phosphide. Notation as in Fig. 3.

Fig. 6. Concentration dependence of E, $\bar{\alpha}_{20-1000}$, λ, ρ in the U–C–N system.

employed for this purpose, and the mechanical properties of the cans were also measured. The test method was set out earlier [19]. The conditions and results of the tests are presented in Table 1.

We see from Table 1 that the carbide of stoichiometric composition does not interact with the can at 800°C through either a helium or a sodium interlayer. The carbide of hyperstoichiometric composition exhibits no interaction in the case of a helium gap, but in the case of a sodium gap carburization of the can occurs. Thus after 4500 h at 800°C an interaction zone of ~100 μ is created. The amount of carbon in the steel can rise to 0.77 wt. %, reducing the ductility by 70%. The characteristic structure of the zone of interaction is depicted in Fig. 7.

In studying the compatibility in direct contact, the most reactive of the compounds studied was naturally uranium carbide. In this case we observed both diffusive penetration of the uranium to a considerable depth and also the separation of carbide inclusions. For the nitride and phosphide there is a very weak interaction at 1000°C, but at 1100°C a liquid phase forms over a very short period in the phosphide.

The behavior of the equimolar uranium carbonitride is analogous to that of the nitride, evidently because of a substantial fall in the activity of the carbon.

An improvement in the compatibility of the carbides of hyperstoichiometric composition may also be achieved by alloying with chromium or zirconium, an increase in the chromium content of the carbide from 5 to 9 wt. % having a favorable effect on the compatibility in the presence of a sodium interlayer. The positive effect of alloying with zirconium was verified for direct contact; there was no interaction at 800°C over a period of 5000 h or at 1000°C for 1000 h.

The interaction of the carbides of hyperstoichiometrical composition with the steel may also be prevented by creating diffusion barriers.

Experiments showed that the use of barriers 20-50 μ thick consisting of copper, chromium, or niobium, or indeed ZrC or TiC, almost entirely prevented the transfer of carbon to the can through the sodium and eliminated interaction in direct contact at 800 or even 1000°C for 1000 h. This period by no means constituted the limiting service period for carbide barriers, the life of which (estimated approximately) sometimes extended to an order of magnitude longer.

The mixed uranium plutonium carbide containing up to 5 vol. % of the sesqui-phase showed a better compatibility through sodium than uranium carbide with an excess dicarbide content. The sesqui-phase was evidently less inclined to give up carbon than the dicarbide. This has been mentioned elsewhere [20].

TABLE 1. Compatibility of the High-Temperature Compounds of Uranium and Plutonium with a Stainless Steel 0Kh16N15M3B Can before Irradiation

Composition	Test conditions			Results of tests			
	medium	temperature, °C	time, h	amount of C in the steel, wt.%	depth of penetration, μ		change in the mechanical properties of the steel
					carbon	uranium	
Uranium carbide (4.8 wt.% C)	Sodium	800	2500	0,06	Not observed	—	No change
The same	Helium	800	2000	0,06	The same	—	The same
Uranium carbide (5.1 wt.% C)	The same	800	2000	0,06	" "	—	" "
The same	Sodium	800	4500	0,77	100	—	ψ fell by 70%
" "	Direct contact	800	2500	Not determined	—	200—250	δ, ψ fell by 30-40% σ_b, $\sigma_{0.2}$ rose by 30-40%
Uranium nitride $UN_{0.92}C_{0.09}O_{0.02}$	The same	800	2500	The same	Not determined	5	No change
The same	" "	1000	2000	" "	The same	10	The same
Uranium phosphide $UP_{1.01}$ (5 wt.% O_2)	" "	800	500	" "	" "	10	σ_b, $\sigma_{0.2}^T$ rose by 10-20% ψ, δ fell by 10-20%
The same	" "	1000	1000	" "	" "	15	σ_b, $\sigma_{0.2}^T$ rose by 10-20% ψ, δ fell by 10-20%
Uranium phosphide $UP_{1.01}$ (5 wt.% UO_2)	" "	1100	2	" "	Formation of a liquid phase	—	No change
Uranium carbonitride $UC_{0.45}N_{0.57}O_{0.02}$	" "	800	2500	" "	Not observed	5	The same
The same	" "	1000	2000	" "	The same	10	" "
Uranium carbide alloyed with chromium (5 wt.%)	Sodium	800	2500	0,1	" "	—	ψ fell by 10%
Uranium carbide alloyed with chromium (9 wt.%)	The same	800	4000	0,8	" "	—	No change
Uranium carbide alloyed with zirconium $U_{0.8}Zr_{0.7}C_{0.98}$	Direct contact	800	5000	Not determined	" "	Not determined	The same
The same	The same	1000	1000	The same	" "	The same	" "
Uranium carbide coated with chromium (5.1 wt.% C)	Sodium	800	4000	0,12	" "	" "	" "
Uranium carbide coated with ZrC	Direct contact	800	2500	Not determined	" "	" "	" "
The same	The same	1000	1000	The same	" "	" "	" "
Mixed uranium and plutonium carbide $(U_{0.8}Pu_{0.2})$ C (5 vol.%) $(UPu)_2C_3$	Sodium	800	2500	0,14	10	" "	ψ fell by 10-20%

We also studied the compatibility of a number of uranium and plutonium compounds with the can under reactor conditions. The results are presented in Table 2. We see from Table 2 that the extent of the layer of interaction varies and depends on the composition of the fuel, the test temperature, and the medium between the fuel and the can. We found that alloying a carbide of hypostoichiometric composition with iron failed to prevent interaction. However, whereas the hypostoichiometric carbide interacts with steel so as to form intermetallic compounds of great hardness, the carbide alloyed with iron only slightly carburizes the can; this is apparently associated with the fact that the carbon released in fission is not combined into a strong carbide at low temperatures.

As in the case of the test carried out before introduction into the reactor, the interaction of the hyperstoichiometric carbide with the can under reactor conditions is far more vigorous than that of the stoichiometric material. Under irradiation uranium carbide and carbonitride interact to roughly the same extent with the can. The layer of interaction has a high degree of hardness, and when the can is strained by ~1% microcracks start forming (Fig. 7b). Sodium is a carbon carrier and accelerates the interaction.

The structure of the can changes throughout its whole thickness (400 μ) although the microhardness changes only slightly and the can has a great reserve of ductility.

TABLE 2. Results of the Interaction of Carbide Fuel with a Kh16N15M3B Steel Can after Irradiation in the BR-5 and SM-2 Reactors

Composition	Conditions of the tests				Depth of penetration, μ	Microhardness of interaction layer, kg/mm²	Character of the interaction
	medium in gap	temperature, °C		burn-up, %, (time, h)			
		average	maximum				
Uranium carbide (4.7 wt.% C)	Helium	570	660	4,5 (6 500)	250	1300	Formation of the liquid phase
Uranium carbide alloyed with iron (4.4 wt.% C; 2.4 wt.% Fe)	The same	435	520	3,8 (6 500)	70	300—500	Carburization
Uranium carbide* (5.0-5.4 wt.% C)	" "	500	570	4,4 (29 300)	40	300—500	The same
Uranium carbide (5.1 wt.% C)	" "	550	690	5,0 (6 500)	140	300—500	" "
Uranium carbide (4.9 wt.% C)	" "	640	740	6,3 (6 500)	100	300—500	" "
Uranium carbonitride (1.6 wt.% C; 3.3 wt.% N)	" "	520	600	4,9 (6 500)	100	300—500	Precipitates along grain boundaries
Uranium carbide (5.1 wt.% C)	Sodium	495	570	3,9 (6 500)	400	270	Carburization
Uranium carbide coated with chromium (5.1 wt.% C)	The same	460	540	3,9 (6 500)	—	290	Absence of interaction

*Samples irradiated in the BR-5 reactor.

TABLE 3. Conditions and Results of the Irradiation of Uranium Monocarbide in the BR-5 and SM-2 Reactors

Serial number	Carbon in UC, wt.%	Maximum burn-up, % of heavy atoms	Temperature in center of fuel (initial), °C	Can temperature (calculated), °C	Swelling for 1% burn-up	Gas evolution, %
1	5,3	0,8	1030	570	1,2	2
2	5,3	2,1	1030	570	1,2	2
3	5,3	3,0	1030	570	1,2	2
4	5,1	3,5	980	500	1,2	—
5	4,8—5,2	4,2	1250	530	2,5	7
6	4,9	6,3	1650	640	3,5	7

Note: Nos. 1, 2, 3, 5, irradiation in the BR-5, Nos. 4, 6, in the SM-2 reactor.

In the case of a helium-filled gap at the same temperature the zone of interaction never exceeded 40 μ.

Coating the uranium monocarbide of hyperstoichiometrical composition with a layer of chromium 25 μ thick prevents the can from suffering carburization on irradiation even when the gap is filled with sodium. The protective layer remains intact for a long time.

Analysis of the results presented in Tables 1 and 2 shows that the character of the interaction under irradiation remains fundamentally the same as under ordinary conditions. However, the velocity of the interaction increases, and there is a considerable carburization of the can at 550-600°C, i.e., 200-250°C lower than in the case of prereactor tests.

Behavior under Irradiation

Irradiation of the high-temperature uranium and plutonium compounds was carried out in experimental assemblies of the SM-2 thermal reactor and the BR-5 fast reactor.

The construction of the fuel elements with carbide fuel was described earlier [19]. The cans of the fuel elements were made of 0Kh16N15M3B stainless steel with a wall thickness of 0.3-0.4 mm. The density of the fuel was 90-93% of theoretical. The porosity laid down in the construction of the fuel element lay in the 20% range and comprised the porosity of the blocks themselves and the diametral can—fuel gap, which varied from 0.18 to 0.4 mm. The gap was filled with a Na—K alloy or helium. The temperature required for the fuel-element can was created by the thermal resistance of the gap between the fuel element and the outer jacket of the experimental ampoule. For measuring the temperatures in the can of the fuel element,

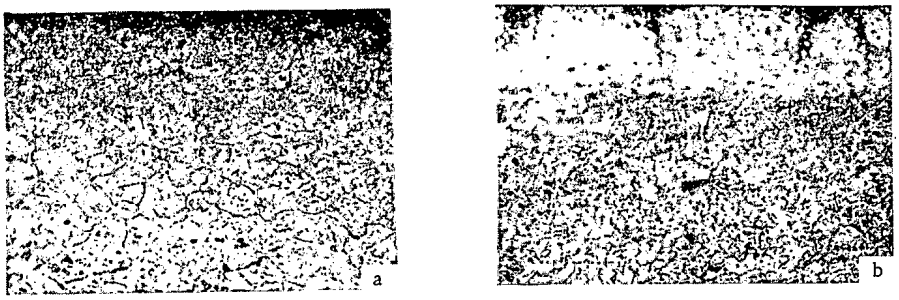

Fig. 7. Microstructure of the zone of interaction of uranium carbide with a can made from 0Kh16N15M3B steel. a) Tests before incorporation into the reactor, 800°C, 450 h, medium sodium; b) tests in the reactor, 550°C, 6500 h, medium helium, can deformed by the swelling of the fuel. Cracks in the carburized layer.

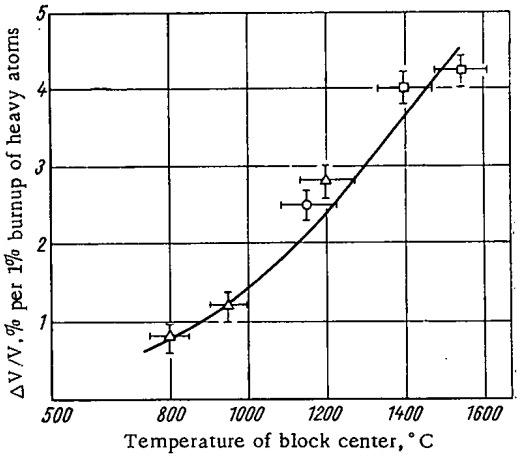

Fig. 8. Rate of swelling of uranium carbide fuel as a function of the temperature in the center of the block. Burnup 3.5-6.3% of the heavy atoms. △○) Free swelling; □) swelling with the cans restrained.

Chromel–Alumel thermocouples 0.2 mm in diameter were introduced into several ampoules. The thermocouples were set in three points over the height of the active section of the fuel element, and normally served for 1.5 years. The thermometric measurements showed that the calculated temperatures were quite close to the real ones. The samples were irradiated at a linear power at 300-600 W/cm to a burnup of 3.5-6.3% of the heavy atoms. The characteristics of certain samples and the conditions of irradiation are shown in Table 3. All the fuel elements remained vacuum-tight after irradiation under the conditions in question. The greatest increase in the diameter of the can in fuel elements with the maximum burnup was about 1%. Investigations showed that the average rate of swelling of the carbide fuel (swelling referred to the burnup averaged over the cross section) depended substantially on the temperature of the fuel (Fig. 8). For a fuel temperature of 700-800°C "hard" swelling in general takes place. The rate of swelling is then 1% for 1% burnup. At 1000-1100°C a considerable contribution is made to the swelling by small blisters less than 0.5 μ in size (Fig. 9). At 1400-1500°C gas bubbles and cavities are formed (Fig. 10) and the rate of swelling increases to 3-4% for 1% of burnup. The number and size of the bubbles and cavities increase from the periphery to the center. In the original state there are only a few pores and they are distributed uniformly over the boundaries and main body of the grains. After irradiation they lie chiefly along the grain boundaries. There are more large pores at the "hotter" boundaries, owing to their migration under the influence of temperature gradients. The inclusions of uranium dicarbide vanish during irradiation. The fact that the swelling of carbide fuel is greater than that of oxide fuel is due to the ability of the carbide fuel to hold gas in the lattice; this is evidently associated with the different diffusion mobility of the gas fragments in uranium dioxide and carbide. The maximum gas evolution from the carbide fuel in the samples under study was 20%. There is also a certain amount of information [1] to suggest that in 10% burnup the gas evolution is about 50%.

X-ray analysis showed that the diffraction maxima remain intact after irradiation in all the samples studied, although the lattice constant of uranium monocarbide diminishes. The greatest diminution in the lattice spacing is 0.16%. A knowledge of the rate of swelling enables us to estimate the degree of porosity, which is important in order to ensure normal working of the fuel element. The introduction of a sodium interlayer into the fuel element between the fuel and the can reduces the temperature of the fuel, which enables us to obtain a burnup of 10% for high thermal loadings. A more difficult problem is that of reaching such a degree of burnup in fuel elements with a gas interlayer between the fuel and the can. According to our own experimental data, such fuel elements are efficient up to a burnup of 6%; however, the can then has an almost critical deformation. The achievement of a higher burnup may clearly develop from a reduction in the effective density of the fuel and also a variation in the distribution of the original porosity.

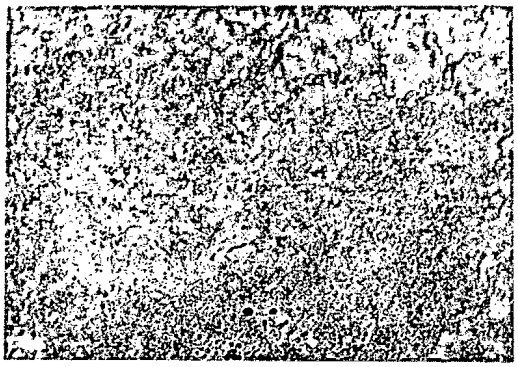

Fig. 9

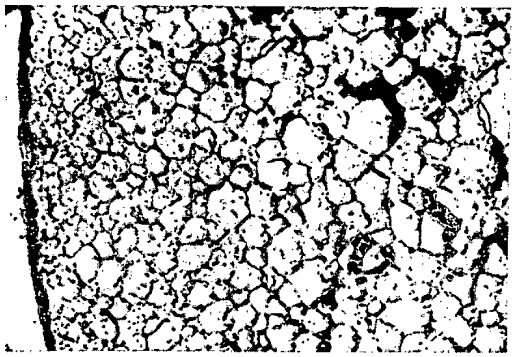

Fig. 10

Fig. 9. Microstructure of uranium carbide after irradiation to a burnup of 4.4% at 1000°C (×7500).

Fig. 10. Microstructure of a core of uranium carbide. Burnup 6.3%, initial temperature in the center of the block ~1600°C (×200).

These and certain other questions which have still to be resolved in order to be able to make use of the prospective forms of high-temperature uranium–plutonium fuel are included in our program for further research in the BOR-60 reactor. For this purpose four groups consisting of 76 fuel elements with carbide cores have been made. The experimental packs will be irradiated to burnup ratios of 3, 5, 7, and 10 wt. %. A detailed study will be made of the effect of various technological arrangements for manufacturing the uranium carbide cores and the composition of the fuel on the stability under irradiation; various constructional arrangements will be studied; the compatibility of the uranium carbide cores with stainless steels in a medium of helium and sodium will also be investigated. The possibility of using fuel elements with a sodium interlayer is extremely attractive. The advantages of carbide fuel may be realized most fully in fuel elements with a sodium interlayer, which by reason of its high thermal conductivity yields high linear powers (1000-1300 W/cm) for a fuel element diameter of 10-12 mm. Higher thermal loadings create impermissibly high thermal stresses in the can. For a helium interlayer the advantages of the carbide fuel are realized most fully for a fuel-element diameter of 6-7 mm. Since the use of large fuel elements in reactors reduces the specific loading, and in the first stages of the development of fast reactors the cost of the plutonium is high, fuel elements with a helium interlayer will find a more extensive use in the first loadings.

Both types of fuel element construction will be studied in the experimental packs: a) elements with a uranium carbide core in a medium of helium; b) elements with a uranium carbide core in a medium of sodium. The behavior of fuel elements with cores made from uranium and plutonium nitrides, sulfides, and other compounds will also be studied.

CONCLUSIONS

1. We have studied the main laws governing the thermophysical and mechanical properties (thermal conductivity, electrical resistivity, specific heat, linear expansion coefficient, hardness, elastic modulus, etc.) of uranium carbide, nitride, carbonitride, sulfide, and phosphide at 20-2000°C.

2. In addition to the generally-accepted methods of producing refractory uranium and plutonium compounds from the dioxide, we have developed a method of heterogeneous gas carbiding and nitriding of the original materials, attractive from the point of view of the simplicity of the apparatus required, the low temperature of the processes, and the possibility of mechanizing and automating the operations involving in the production of fuel-element cores.

3. We have studied the compatibility of refractory uranium and plutonium compounds with 0Kh16-N15M3B steel before introduction into the reactor and also under reactor conditions. The character of the interaction is exactly the same, but the temperature at which UC starts interacting with the can is 200-250°C lower under irradiation. In the case of carbide fuel the interaction expresses itself by carbiding the can to a depth of 100-400 μ, although this fails to rupture the fuel elements. The compatibility may be improved by depositing protective coatings on the core (chromium, titanium, zirconium, etc.).

4. Fuel elements with carbide and carbonitride fuel have been irradiated without rupture to a burnup of 6.3% of the heavy atoms at a linear power of up to 600 W/cm and a temperature of 1600°C in the center. The swelling of the carbide fuel depends on the temperature. Gas evolution from carbide fuel reaches 20% for a burnup of 6.3%.

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RADIATION SAFETY IN THE USSR

A. I. Burnazyan, S. M. Gorodinskii,
L. A. Il'in, V. M. Kozlov,
A. N. Marei, G. M. Parkhomenko,
and A. D. Turkin

The protection of radiation workers and the general population in the USSR against radiation is ensured by the following system of governmental measures:

1) the establishment of legal standards regulating the dose to individuals working with sources of ionizing radiation and to the general population;

2) the development of a system of protective regulations covering individuals and groups directed toward the assurance of safe working conditions (for example, the promulgation of health rules for all branches of industry to prevent overexposure of individuals and the general population);

3) the creation of a system of State health surveillance for compliance with regulations in this field, and the establishment of a medical service for supervising the health of workers;

4) the organization of scientific studies aimed at the prevention of radiation effects on the health of individuals and of the general population.

The basic biological effects of ionizing radiation on the body have been established by extensive experimental studies and by clinical observations. A particularly large amount of effort has been devoted to studies of such problems as the kinetics of the accumulation and decomposition of radioactive materials, the reactions of the body and the biochemical changes occurring in it because of the effects of ionizing radiation, and the disturbance of the functional state of tissues and organs. These experimental and theoretical studies have proved to be extremely fruitful in the development of maximum permissible levels of ionizing radiation and maximum permissible intake of radioactive materials into the body.

It should be emphasized that the maximum permissible levels of irradiation changed significantly with the accumulation and refinement of knowledge about the effects of ionizing radiation on the body. Where the permissible dose of γ-radiation was 0.1 R per working day in 1946-1950, it was reduced to 0.05 R, and then to 0.017 R in succeeding years.

The radiation safety standards (NRB-69) now in effect in the Soviet Union are mainly based on the recommendations of the International Commission on Radiation Protection (ICRP).

Radiation safety standards in the USSR have the force of law. They are distributed to establishments, institutions, laboratories, and other organizations in all ministries and departments.

Work with radioactive materials and other sources of ionizing radiation is carried on with the permission of agencies of the State health surveillance group, to which is presented all the information about radiation levels and environmental contamination necessary for evaluation of radiation hazard.

Units of the State health surveillance group accomplish their purpose by examination of plans, inspection during the course of construction, and participation in the acceptance of supplies.

The health rules issued by the USSR Ministry of Health are binding on all departments. They regulate the conditions for the location and operation of nuclear installations which are potential sources of radiation effects for workers and the general population. In addition to the usual health rules for working with

Ministry of Health, USSR. Translated from Atomnaya Énergiya, Vol. 31, No. 4, pp. 403-409, October, 1971.

TABLE 1. Recommended Minimum Protection Factors for Personnel Protection Methods for Radioactive Aerosols

Personnel protection method	Minimum protection factor
Supplied-air personnel protection methods (suits, jackets, and masks)	10 000
Protective suits with self-contained regeneration system for expired air	1000
Antiaerosol respirators	100

TABLE 2. Standards for Personnel Protection Methods Defining Functional State of the Body

Personnel protection method	Air supply rate, liter/min	Respirator resistance at constant air flow of 30 liter/min, mm water	Wt., kg
Supplied-air personnel protection methods: suits and sets for gas and electric welders	250—400	10	6
jackets	200—350	10	2
helmets and masks	150—200	10	0,6
Protective suits with self-contained regenerative system for expired air	—	30	7
Antiaerosol respirators	—	3	0,3

radioactive materials and other sources of ionizing radiation, there are special health rules in the USSR (for example, health rules covering the design and operation of atomic power stations, rules for the transportation of radioactive materials, etc.). Surveillance for compliance with radiation safety standards and health rules is accomplished by health units which are radiologic subsections of the health stations in the Ministry of Health. Workers in the health stations are given extensive powers — up to the shutting down of an offending unit. Along with the health surveillance, day-to-day monitoring for compliance with radiation safety standards is carried on in the units. The day-to-day monitoring is carried out by a radiation safety service (RSS) which is a direct part of the structure of an enterprise. The RSS of an enterprise provides constant observation of radiation equipment and production lines, of the amount of radioactive waste discharged into the surroundings, and of contamination levels in environmental objects (air, reservoirs, food products, etc.). The RSS informs higher-level departments and units of the State health surveillance group about the results of its observations, particularly in the case of an observed violation of health regulations, and works together with them to eliminate the observed violation.

In research institutes, group and individual means for protection of personnel against radioactive gases and aerosols, new methods of personnel dosimetry, and methods for lifetime determination of radioactive isotopes in the body are being investigated; studies are being made of rational arrangement and grouping of equipment, of planning for working areas, and of the reasons for general and industrial disease. Industrial hygiene studies make possible the accumulation of important material on health evaluations of equipment used, protective devices, everyday health facilities, and protective covers on structural units. In contrast to the requirements imposed on equipment in other branches of industry, in working with radioactive materials and particularly high-activity materials, there is good reason to require the sealing not only of individual parts of the equipment and individual chambers (boxes) but also of the entire process line with complete elimination of the removal of radioactive materials and of parts contaminated by them outside the confines of the box system. The development of special zonal planning of areas is something essentially new. It should be pointed out that special boxes equipped with gloves and manipulators have come into widespread use for ensuring safety in working with radioactive materials.

An important position is occupied by a system of protective locks and protective pass-throughs which ensured localization of radioactive contamination in the area where the work is being performed.

Theoretical premises about the use of zonal division of production areas and requirements imposed on equipment and ventilating facilities on the basis of industrial hygiene studies are now increasingly confirmed in practice and are being used in planning installations. In addition to the work on industrial hygiene aspects of working conditions, the operational developments in the determination of radioactive materials in various media are worthy of note.

The principle directions of the scientific studies being carried out in the field of population protection against the effects of ionizing radiation are the characteristics and health aspects of radiation sources, the analysis of radioactive isotope behavior in the various portions of the environment and the establishment of the paths by which interaction with humans occurs, the determination of relationships between the content of radioactive materials in the environment and in the body, the study of the effect of ionizing radiation on living conditions and health of the population, the prediction of the radiation environment and of possible health consequences for the population because of the effects of ionizing radiation, the development of health measures directed toward the assurance of effective protection of the population against the effects of ionizing radiation.

TABLE 3. Uranium and Radium Content in Discharge Water and in Reservoirs

Isotope	Mine waters		In reservoir water at waste water discharge point
	before purification	after purification	
Natural uranium, mg/liter	0,2—0,6	0,1 and less	0,003—0,1
Ra^{226}, Ci/liter	10^{-11}—$5 \cdot 10^{-10}$	$(0,5 \div 8) \cdot 10^{-11}$	$(0,4 \div 2) \cdot 10^{-11}$

A knowledge of the composition of radioactive wastes and of the properties which determine their characteristics, of the behavior of the corresponding isotopes in reservoirs and of the nature of their migration along the food chain as well as of the frequency of various levels of accumulation in the bodies of different population groups makes it possible to justify and include in the health regulations the appropriate requirements for the disposal of liquid radioactive wastes. The use of recirculating water supply after the discharge water is cleared of radioactive contamination offers a practical possibility for solving the problem of health protection for reservoirs with respect to contamination by liquid wastes from nuclear installations.

For the health protection of underground waters, studies were made of the mechanisms and relationships which determine the migration of radioactive isotopes in various soils and geologic formations. This made it possible to develop methods for predicting the hydrologic conditions for the entrance of radioactive materials into water-bearing horizons; the studies also furnished a scientific basis for inclusion of requirements in the health regulations which controlled the conditions for storage of solid and liquid wastes.

In performing repair and emergency operations, there is always the possibility of air contamination; consequently, a high priority was assigned to the creation of methods for individual personnel protection.

Since 1953, supplied-air protective suits (pneumosuits) with ventilation of the entire space within the suit with pure air have been used as a means of individual protection of skin and respiratory organs during the most severe repair, decontamination, and emergency operations. High protective efficiency, low physiologic load, simplicity, and operational reliability are the basic characteristics of the LG-U and LIZ-KS pneumosuits. The minimum protective factors of these suits against aerosols are $1.3 \cdot 10^4$ and $5 \cdot 10^3$, respectively.

For repair and emergency operations requiring high mobility, a protective suit with a self-contained air supply was designed in which there was regeneration of the air by means of alkali metal peroxides, circulation of air by using the space inside the suit as a breathing bag, and thermoregulation of the body by external cooling. The minimum protective factor of this suit against aerosols is $3.5 \cdot 10^3$.

Along with the creation of supplied-air masks and helmets, there was developed the "Lepestok" antiaerosol valveless disposable respirator. The use of the ultrathin-fiber, electrostatically charged filtration material FPP as housing filter and obturator and the absence of a valve housing made it possible to construct a set of cheap respirators with low respiratory resistance and high protective efficiency against aerosols of different dispersions: "Lepestok-200," "Lepestok-40," and "Lepestok-5."

To protect the skin in repair areas and to prevent scattering of radioactive contamination outside the repair area, a system was devised using special temporary-use plastic clothing and additional booties which are worn over the special clothing and booties used daily.

As the result of a study of the features of skin contamination and decontamination with respect to various radioactive materials, the decontaminating agents "Zashchita," "116," and others were developed and used successfully in practice. The problem of skin decontamination can now be considered as solved in the USSR as far as radioactive materials are concerned.

The creation of new means of individual protection for operations involving radioactive materials gave rise to a need for investigations into the nature of the decontaminability of materials and the interaction of those materials with contaminating and decontaminating agents, for a study of the protective properties of personnel protection agents against radioactive isotopes in various combined states, and for the development of standards for individual protection methods.

TABLE 4. Personnel Irradiation at Commercial Reactors in 1969

Radiation dose, rem	Number of people working at commercial reactors, %	Radiation dose, rem	Number of people working at commercial reactors, %
To 1,0	54	2,6–5	13
1,1–2,5	33	5,1 and above .	—

TABLE 5. Dose Composition for Personnel Irradiation at Commercial Reactors, Averaged over a Year, %

β-Radiation	γ-Radiation	Thermal neutrons	Entire neutron spectrum
7–20	70–80	5–7	10–15

In working with radioactive materials, an extensively used special material is one having a polyvinylchloride base into which is introduced an unbound paraffin additive that forms a continuously released thin layer on the surface of the material. This inert nonwetting layer prevents penetration of the majority of radioactive contaminants into the body of the material.

Specific recommendations were also developed for methods of decontaminating special clothing and other personnel protection devices which take into account the nature and state of the radioactive contamination.

From the large arsenal of methods for evaluating protective efficiency, perference is given to direct methods. Since the beginning of the 1960's, direct methods were used to evaluate protective suits, supplied-air helmets, respirators, and other means of individual protection by means of simulated radioactive aerosols.

Physiological studies of personnel protection methods are also of great importance.

Studies of physiological changes in the bodies of people using personnel protection methods make it possible to recommend the appropriate standard indicators for gaseous impurities in inspired air which are formed in the body and which may accumulate when using some self-contained systems for the regeneration of respiratory mixtures.

On the basis of these physiological studies and studies of operating experience with protective measures in the Soviet Union, standards were developed for personnel protection methods not only with respect to protection efficiency (Table 1) and harmful impurities in respiratory mixtures, but also with respect to other indicators which define the functional state of an individual using the personnel protection methods. The most important of these requirements are given in Table 2.

Great advances in the health conditions of working people and in living conditions for the general population have been achieved as a result of all the studies that have been completed and of the strict inspection system for compliance with regulations and radiation safety standards.

As examples, one can discuss the working conditions and radiation estimates in the neighborhood of uranium mines, industrial and power reactors, and radiochemical enterprises.

During the early operations of the uranium mining industry, the dominating factor in the effects on miners was the quartz-containing dust which produced silicosis. Because of this, efforts were chiefly directed at reduction of dust concentrations in the air of mine workings by the introduction of wet drilling, measures for hydraulic dust removal during explosive operations, the use of dust-wetting additives in wash water, etc; overall mine ventilation was improved because of the introduction of varied ventilating equipment and systems. As a result of these measures, significant results in the reduction of mine air dustiness to maximum permissible concentrations and less were achieved even in the 1950's. While the MPC for dust loading in air with free silicon dioxide content above 10% is 2 mg/m^3, the weighted mean dust loading was reduced to levels of 0.7-1.4 mg/m^3 in the 1950's in all uranium mining enterprises. This ensured the elimination of silicosis cases among miners with the exception of individual cases of the disease in people who came to work in the 1940's and early 1950's.

Because of increased radon concentrations in the air in underground workings, the development and introduction of antiradon measures was started in the early 1950's. A method for designing mine ventilation including consideration of radon was developed. A whole series of antiradon measures were introduced in the form of ventilation improvement both generally throughout the mine and at working faces, isolation and sealing of exhausted or temporarily inactive workings, deposition of antiradon coverings on emanating surfaces of mine workings, capping of mine water, etc. Through the realization of the antiradon measures, the radon concentration in the air of underground workings at the end of the 1950's was reduced to permissible levels.

TABLE 6. I^{131} Irradiation and Contamination Levels Requiring Population Protection Measures in Emergency Situations*

Quantities specified in recommendations or used in calculations	USSR	Great Britain	USA
Radiation dose to the thyroid, rad	25/250†	25	30(10)‡
Intake with food, μCi:			
children	0.8/8	1.7	1.75 (0.58)
adults	13.5/135	15	—
Intake by inhalation, μCi:			
children	1.1/11	2.2	—
adults	18/180	20	—
Maximum contamination of food (f, μCi/day) or milk (m, μCi/day):			
children	0.95/9.5	3 (m)	—
adults	0.06/0.6 (f, m)	0.25 (m)	0.25(0.84)(m)
Air concentration, nCi/liter:			
children	3/30	0.6	—
adults	20/200	24	—

*Values underlined are given in the recommendations.
† The first number is level A, the second is level B.
‡It is understood that irradiation of any individual will not exceed 30 rad if the average dose to the corresponding population group does not exceed 10 rad.

As is clear from what has been said, great advances in terms of radical improvements in health conditions at work have been achieved during the period the uranium mining enterprises were in operation.

However, considering that the intake of even minimum amounts of long-lived radioactive isotopes into the body is undesirable, particularly because of possible combined effects with other industrial factors, all steps are being undertaken in the mines so that the greatest possible further reduction of dust loading and radon content in mine air is achieved.

As a rule, uranium aerosols are not detected in the atmosphere at populated points near the mines, and the radon concentration is at a level which is typical for the locality.

Mine waters are a second potential source of environmental contamination. The water undergoes purification (by the sorption method) in appropriate devices before discharge into a river at enterprises having large amounts of mine water. One can judge the purification efficiency from the data in Table 3. So low a concentration of the indicated isotopes in purified mine waters and in reservoir water satisfies health requirements.

For all operations the uranium air concentration for mining departments is 0.01-0.02 mg/m^3, and for metallurgical departments, 0.07-0.2 mg/m^3 compared to an MPC of 0.2 mg/m^3. The radon content in air is within the limits $(1-3)\cdot10^{-11}$ Ci/liter while the MPC is $3\cdot10^{-11}$ Ci/liter.

Hydrometallurgical factories also present no hazard as sources of atmospheric contamination. It is sufficient to point out that the average Ra^{226} and Po^{210} escaping with industrial discharges is considerably below the MPC in the air of nearby populated sites.

Semiliquid radioactive wastes are held in liquid waste storage reservoirs. The filtered water from them is returned to process lines. Excess water is purified to established standards and discharged into a reservoir.

In plants producing metallic uranium and manufactured uranium parts, the discharges containing uranium aerosol undergo special purification. As a result, the uranium air concentration in inhabited settlements is $(0.6-1.0)\cdot10^{-5}$ mg/m^3, i.e., tens and hundreds of times less than established standards. In the health protection zone, the α-emitter content in soil, vegetation, and snow remains within the range of the natural background.

This applies equally to the water discharged by plants of this type. The Ra^{226} and natural uranium content in water and in the bottom layers of reservoirs receiving plant discharge water are at the level of the natural background, i.e., there is practically no contamination of the reservoirs by radioactive isotopes.

At commercial uranium–graphite reactors, external γ and neutron radiation are the main factors involved in radiation effects. The annual radiation dose to personnel operating commercial reactors at the end of the 1950s was: up to 5 R, 95% of all personnel; 5-10 R, 4.7%; up to 15 R, 0.3%. These numbers did not exceed the maximum permissible values in the USSR at that time. To improve the radiation environment in subsequent years, attention was directed toward the prevention of emergency situations and toward the reduction of repair operation volume by increased reliability of technical equipment.

Because of these measures, personnel irradiation at industrial reactors basically does not exceed 2.5 rem/year with the exception of individuals concerned with repair and emergency operations, for whom the radiation dose is somewhat higher but not more than 5 rem/year (Tables 4 and 5).

Proper arrangement of shielding, zonal design of working areas, and remote control of technical processes provided safe working conditions in the radiochemical industry. Over the last seven years, more than 80% of the personnel had an annual dose less than 1 R and only 2% received from 2.5 to 5 R.

Measurements of the content of γ-emitting isotopes in the bodies of workers at commercial reactors and radiochemical factories made with whole-body counters show that the activity in the body does not exceed 10^{-7} Ci in the overwhelming majority of cases. Cs^{137} in amounts up to $8 \cdot 10^{-7}$ Ci in the body was observed only in particular individuals. The presence of Cs^{137} and Zn^{65} in the body was typical of workers at commercial reactors; the presence of Cs^{137} and Zr^{95} together with Nb^{95} was typical of workers at radiochemical factories. The thyroid content of I^{131} was $5 \cdot 10^{-10}$-$3 \cdot 10^{-9}$ Ci. The data presented indicate the content of radioactive isotopes in the bodies of workers at commercial reactors and radiochemical plants is considerably less than the maximum permissible content recommended by the ICRP for the isotopes mentioned.

Environmental contamination by radioactive materials around industrial reactors and plants for radiochemical processing of irradiated uranium is insignificant. The doses of localities outside the boundaries of the health protection zone (5 km) for these plants (downwind) did not differ from the doses at control points and did not exceed 100-120 mrem/year. In the region of the closest populated site (a distance of 8 km), there was practically no increase over the natural γ-background, and the content of long-lived aerosols in the atmosphere did not exceed the level of global fallout. The specific activity in discharge water was $8 \cdot 10^{-9}$-$3 \cdot 10^{-8}$ Ci/liter; the activity resulted mainly from the presence of Mn^{56}, Na^{24}, and partly from P^{32}. The Sr^{90} concentration was no more than 10^{-10} Ci/liter.

Limited volumes of discharge water containing long-lived isotopes with a specific activity of 10^{-4}-10^{-3} Ci/liter are pumped into deep water-bearing horizons. A small amount of high-activity water containing long-lived isotopes is stored in special tanks.

Operating experience at the Beloyarsk, Novo-Voronezh, and other atomic power stations indicates that the radiation environment in working areas is completely satisfactory. Personnel irradiation levels at an atomic power station are considerably below 5 rem/year; the discharge of inert radioactive gases into the atmosphere is 100-500 Ci/day, of aerosols, no more than 10^{-3} Ci/day, and of I^{131} no more than 0.05 Ci/day.

Many years of experience in the operation of atomic industrial and power plants enables us to state that the radiation environment, both within a plant and in the surrounding area, is completely favorable under normal conditions. At the same time, emergency situations, in which personnel overexposures are possible, bring up the problem of prevention and elimination of radiation accidents. Criteria have been developed for making emergency decisions in accident situations – one of the most important facets of this problem. As an illustration, we can discuss an emergency situation associated with areal contamination by I^{131} – one of the most dangerous isotopes in the accidental discharge into the atmosphere of radioactive materials from a reactor.

The "Criteria for Making Emergency Decisions" effective in the USSR, in which radiation and contamination levels are defined, provide the basis for determining the need to take definite measures to protect the population or the absence of such a need. The decision is made on the basis of a comparison of estimated (predicted) danger levels with levels A and B in the criteria (Table 6).

Level A. If the hazard does not exceed level A, there is no need to take extreme measures which would cause temporary disruption of the normal activities of the population.

Level B. If the hazard reaches and exceeds level B, it is necessary to take extreme measures even if they lead to temporary disruption of the normal activities of the population and of the economy of the region.

If the hazard exceeds A but does not reach B, the appropriate decision is made on the basis of circumstances and local conditions. Because the criteria contain more than a single guideline, they permit a certain flexibility in reaching decisions depending on specific circumstances, and at the same time they make it possible to avoid serious over- or underestimation of the danger. These criteria are a component part of the program for radiation protection of the USSR population in emergency situations.

RADIOACTIVITY OF THE WATERS OF THE WORLD OCEANS AND BEHAVIOR OF SOME FISSION ELEMENTS IN THE OCEANS

V. M. Vdovenko, A. G. Kolesnikov,
V. I. Spitsyn, R. N. Bernovskaya,
L. I. Gedeonov, V. V. Gromov,
L. M. Ivanova, B. A. Nelepo,
V. N. Tikhomirov, and A. G. Trusov

Radioactive wastes are nowadays increasingly discharged into water. The importance of the discharge processes for the radioactive contamination of the ocean became in certain locations comparable to the importance of the global radioactive fallout. A bibliography of the radioactive contamination of the ocean has been given in comprehensive form in [1, 2]. Of particular interest are [3, 4] among the more recent publications. The present article reports the latest results of research whose previous stages were outlined in [5-7].

The global radioactive contamination of the ocean was basically derived from observation data concerning the concentration of the long-living radioactive isotopes Sr^{90} and Cs^{137} [8, 9]. The present article describes the results of our recent measurements. In investigations of the radioactive contamination caused by the discharge of radioactive wastes one must keep track of a series of radioactive isotopes whose half-life periods are shorter than those of Sr^{90} and Cs^{137} and amount to months and days. Important information can be obtained from the isotopes Ce^{144}, Y^{91}, Nb^{95}, and Mo^{99} (Tc^{99}). The results of first stages of research on the behavior of these isotopes in sea water are listed below.

Table 1 lists the concentration of Sr^{90} and Cs^{137} in the surface water at 92 points of the Pacific at the end of 1966 and the beginning of 1967 and at some points of the Atlantic ocean in 1967. All data of Table 1 are indicated on the maps shown in Figs. 1 and 2. Table 2 lists the depth distribution of Sr^{90} and Cs^{137}. Reliable results were obtained at all observation points down to depths of 500 m. The concentrations at greater depths were frequently below the detection limit of the instruments. However, there were places at which at depths of 1000 m or more, the Sr^{90} and Cs^{137} concentrations were above the detection limit and could be determined with adequate reliability.

The Sr^{90} and Cs^{137} quantities stored in the water layer between the surface and a depth of 500 m were estimated at many points, and at depths as great as 1000 m at a few other points. The calculations were based on the depth-dependence of the concentration.

Table 3 lists the corresponding results. The accumulations calculated are indicated on the maps of Figs. 3 and 4. Several peculiar features of the radioactive contamination of the water in certain regions of the world ocean were noticed.

At one point in the Sea of Japan, the Sr^{90} concentration amounted to 71 decays/min/100 liters of water in November, 1966 and the corresponding Cs^{137} concentration resulted in 79 decays/min/100 liters of water. A comparison of these figures with the data obtained in the Mediterranean and the Black Sea [5 6] reveals that the concentrations are in all closed seas higher than in the open seas (the same latitudes were considered). The global radioactive fallout decreased from 1962 to 1966 [10], but the Sr^{90} and Cs^{137} concentrations in these seas decreased only insignificantly. The respective Sr^{90} and Cs^{137} quantities stored in the Sea of Japan amounted to 95 and 117 mCi/km^2 in the layer extending from the surface to a depth of

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TABLE 1. Sr^{90} and Cs^{137} Concentrations in the Surface Water of the Oceans

Date of sampling	Coordinates of point at which sample was taken		Concentration (decays /min · 100 liter)		Cs^{137}/Sr^{90}
	latitude	longitude	Sr^{90}	Cs^{137}	
Pacific Ocean					
1966					
1/XI	42°19′N	134°32′E	71±11	79±11	1,1
5/XI	40°00′N	142°15′E	44±7	66±9	1,5
6/XI	36°00′N	142°41′E	46±7	70±11	1,5
8/XI	30°05′N	144°07′E	45±7	38±10	1,9
9/XI	25°16′N	145°17′E	36±5	70±8	1,9
10/XI	22°30′N	145°46′E	43±6	81±11	1,8
11/XI	19°36′N	147°25′E	49±7	56±10	1,1
13/XI	13°59′N	147°08′E	43±7	52±7	1,2
14/XI	13°40′N	149°46′E	18±3	33±3	1,8
15/XI	13°22′N	152°24′E	23±4	44±6	1,9
16/XI	12°55′N	154°47′E	21±3	41±6	1,9
17/XI	12°35′N	158°72′E	22±4	42±6	1,9
18/XI	11°42′N	161°30′E	22±4	38±5	1,7
19/XI	12°08′N	162°10′E	30±5	56±14	1,8
20/XI	12°05′N	164°46′E	21±3	40±9	1,9
21/XI	12°01′N	167°13′E	14±3	34±6	1,7
22/XI	10°33′N	170°55′E	10±2	15±5	1,5
23/XI	8°29′N	171°15′E	19±3	31±9	1,6
24/XI	4°43′N	169°45′E	38±6	42±11	1,1
25/XI	0°09′N	167°11′E	31±5	39±14	1,2
26/XI	3°22′S	165°00′E	20±3	27±11	1,3
27/XI	7°06′S	162°56′E	27±4	32±11	1,2
29/XI	13°26′S	158°42′E	9±2	10	—
30/XI	16°51′S	157°30′E	11±2	11±2	1,3
1/XII	21°10′S	157°22′E	12±2	21±8	1,7
2/XII	25°18′S	157°10′E	10±2	—	—
3/XII	28°59′S	154°46′E	17±3	28±9	1,6
4/XII	32°53′S	152°20′E	22±4	32±13	1,4
5/XII	32°53′S	152°20′E	26±4	30±12	1,1
10/XII	30°48′S	153°22′E	27±4	37±15	1,4
11/XII	33°29′S	158°30′E	11±2	13±7	1,2
12/XII	33°01′S	165°04′E	10±2	16±7	1,6
13/XII	32°41′S	169°19′E	8±2	—	—
14/XII	32°20′S	174°26′E	9±2	—	—
15/XII	32°05′S	175°10′E	10±2	—	—
16/XII	32°03′S	177°20′W	9±2	—	—
16/XII	32°03′S	177°20′W	1±2	10	—
18/XII	28°43′S	178°06′W	9±2	16±7	1,7
19/XII	22°57′S	174°35′W	9±2	10	—
21/XII	20°13′S	173°10′W	30±5	—	—
23/XII	19°39′S	169°02′W	55±8	—	—
25/XII	18°47′S	160°34′W	130±25	—	—
26/XII	17°57′S	153°03′W	26±4	26±9	1,0
27/XII	17°20′S	152°40′W	19±3	—	—
30/XII	18°03′S	146°45′W	31±5	—	—
30/XII	18°43′S	144°54′W	13±2	22±8	1,7
31/XII	19°25′S	142°32′W	10±2	17±7	1,7
1967					
1/I	20°14′S	140°24′W	12±2	18±9	1,5
2/I	19°50′S	138°27′W	13±2	15±8	1,1
4/I	16°05′S	137°36′W	9±2	16±6	1,7
5/I	11°40′S	135°43′W	7±1	13±7	1,8
6/I	9°45′S	135°25′W	10±2	—	—
7/I	5°47′S	135°43′W	8±1	—	—
8/I	1°12′S	134°59′W	9±2	—	—
9/I	0°34 N	134°35′W	10±2	—	—
10/I	4°35′N	133°15′W	10±2	—	—
11/I	9°07′N	132°07′W	12±2	—	—
12/I	13°14′N	131°04′W	17±3	—	—
13/I	15°53′N	130°26′W	17±3	—	—
14/I	20°10′N	120°04′W	24±4	—	—
15/I	23°41′N	127°42′W	34±5	—	—
16/I	28°08′N	126°20′W	63±10	—	—
17/I	32°10′N	125°13′W	65±10	—	—
24/I	38°27′N	125°07′W	39±6	—	—
25/I	41°15′N	126°36′W	178±25	—	—
25/I	43°32′N	127°58′W	126±19	100±12	0,8
26/I	41°18′N	126°06′W	106±16	119±12	1,1
26/I	44°37′N	125°37′W	90±14	123±14	1,3
27/I	45°13′N	125°07′W	65±10	110±12	1,6
27/I	45°32′N	124°51′W	64±10	98±10	1,5
27/I	46°14′N	124°43′W	60±10	89±10	1,5
28/I	46°35′N	124°35′W	55±8	76±10	1,4
28/I	46°52′N	124°28′W	45±7	73±10	1,6
29/I	46°41′N	125°45′W	69±10	102±12	1,5
29/I	46°32′N	126°28′W	75±11	107±12	1,4
29/I	45°37′N	126°05′W	50±8	100±11	2,0
29/I	44°39′N	125°52′W	51±7	90±12	1,7
30/I	43°10′N	125°15′W	62±9	106±13	1,7
31/I	39°16′N	123°58′W	39±6	—	—
1/II	35°07′N	122°43′W	62±9	64±9	1,0
2/II	31°14′N	120°15′W	46±7	—	—
3/II	28°08′N	116°48′W	57±9	118±13	2,0
4/II	25°01′N	113°31′W	47±7	—	—
5/II	22°13′N	110°16′W	39±6	—	—
6/II	20°33′N	107°41′W	31±5	—	—
7/II	18°19′N	103°56′W	16±3	—	—
8/II	16°21′N	99°39′W	25±4	—	—

Date of sampling	Coordinates of point at which sample was taken		Sr^{90} decays /min · 100 liter
	latitude	longitude	
9/II	14°15 N	95°50′W	14±2
10/II	12°12 N	92°19′W	10±2
11/II	10°58 N	83°56′W	15±3
12/II	9°09 N	85°14′W	19±3
13/II	6°58 N	81°47′W	14±2

Date of sampling	Coordinates of point at which sample was taken		Sr^{90} decays /min · 100 liter
	latitude	longitude	
Atlantic Ocean			
1967			
16/II	11°24′N	78°10′W	9±2
17/II	14°25′N	80°29′W	14±2
18/II	19°04′N	80°10′W	21±3
19/II	18°58′N	82°09′W	5±1
20/II	20°51′N	84°40′W	7±1
22/II	22°17′N	87°37′W	10±2
23/II	21°17′N	92°03′W	20±3
2/III	25°00′N	90°50′W	9±2
3/III	27°12′N	88°41′W	3±1
4/III	26°06′N	86°50′W	3±1
5/III	24°23 N	84°15 W	19±3
8/III	24°24′N	80°48 W	17±3
9/III	27°46′N	79°49′W	6±1
9/III	27°50′N	79°42′W	17±3
10/III	28°04′N	79°31′W	22±4
11/III	29°36′N	78°45′W	19±3
12/III	29°58′N	77°55′W	15±2
15/III	28°42′N	73°37′W	7±1
16/III	26°47′N	67°56′W	6±1
17/III	24°30′N	65°09′W	5±1
19/III	22°55′N	61°48′W	32±5
20/III	20°28′N	57°16′W	21±3
21/III	17°30′N	50°01 W	8±1
22/III	14°00′N	52°14′W	4±1
23/III	11°19′N	50°19′W	3±1
24/III	8°57′N	48°17′W	14±2
26/III	2°51′N	43°12′W	17±3
27/III	1°00′N	41°58′W	11±2
2/IV	0°20′S	37°28′W	3±1
3/IV	1°31′N	34°56′W	21±3
4/IV	0°27′N	34°25′W	12±2
5/IV	4°32′S	34°25′W	7±1
6/IV	8°42′S	34°32′W	10±2
7/IV	12°25′S	36°36′W	16±3
8/IV	15°58′S	37°27′W	5±1
9/IV	19°44′S	39°05′W	5±1
11/IV	25°19′S	41°45′W	10±2
19/IV	22°38′S	41°15′W	8±1
21/IV	19°08′S	33°41′W	9±2
22/IV	16°52′S	29°43′W	5±1
25/IV	10°02′S	18°58′W	4±1
26/IV	7°32′S	15°27′W	7±1
27/IV	4°42′S	11°57′W	5±1
28/IV	2°32′S	8°29′W	7±1
29/IV	0°07′N	4°48 W	5±1
2/V	00°00′	00°00′	6±1
4/V	2°19′N	4°43′W	3±1
6/V	6°45′N	12°40′W	9±2
7/V	9°46′N	16°20′W	11±2
8/V	13°16′N	17°36′W	6±1
10/V	21°26′N	17°39′W	3±1
12/V	30°34′N	11°25′W	16±3
12/V	30°34′N	11°25′W	10±2
12/V	30°34′N	11°25′W	6±1
16/V	36°16′N	7°54′W	6±1
18/V	42°42′N	4°33′E	8±1
18/V	42°45′N	5°14′E	3±1
25/V	36°39′N	20°57′E	22±4

TABLE 2. Sr^{90} and Cs^{137} Concentration in the Deep-Sea Water of the Oceans

Date of sampling	Coordinates	Depth, m	Depth at which the sample was taken	Concentration, decay/min · 100 liter		Cs^{137}/Sr^{90}
				Sr^{90}	Cs^{137}	
Pacific Ocean						
1966						
1/XI	42°13′N 134°32′E	3700	0	71±11	79±11	1,1
			200	40±6	55±8	1,3
			500	26±5	27±7	1,0
			1000	7±3	10±5	1,4
6/XI	36°00′N 142°41′E	8412	0	46±7	70±11	1,5
			500	14±3	—	—
			1000	8±2	10	—
			2000	5	10	—
10/XI	22°30′N 145°46′E	7673	0	43±7	81±11	1,8
			500	19±3	27±10	1,4
			1000	19±3	21±8	1,1
			5000	6±1	11±6	1,8
			7000	10±2	17±6	1,7
11/XI	19°36′N 147°25′E	6659	0	49±7	56±10	1,1
			200	37±6	48±8	1,3
			500	11±2	10	—
			1000	11±2	10	—
			2000	5	10	—
			3000	5	—	—
			6000	5	10	—
13/XI	13°59′N 147°08′E	9295	0	43±7	52±7	1,2
			100	39±6	66±9	1,7
			200	18±4	21±5	1,1
			500	9±2	10	—
			1000	—	—	—
			8000	5	10	—
18/XI	11°42′N 161°30′E	1940	0	22±4	42±6	1,9
			150	22±4	29±3	1,3
			500	10±2	—	—
			1000	7±2	—	—
19/XI	12°08′N 162°10′E	2850	0	30±6	56±14	1,8
			500	15±3	20±7	1,3
			1000	11±2	14±6	1,3
			2000	17±3	29±9	1,7
20/XI	12°05′N 164°46′E	1587	0	21±3	40±9	1,9
			200	9±2	16±7	1,8
			500	9±2	17±8	1,9
			1000	6±2	10±5	1,6
			2000	5±3	10	—
25/XI	0°09′N 167°11′E	4 5 2	0	31±5	39±14	1,2
			500	11±5	17±8	1,5
			1000	7±3	12±6	1,7
			2000	5	10	—
			4000	5	10	—
29/XI	13°26′S 159°42′E	3 890	0	9±2	10	—
			500	5±2	10	—
			1000	6±2	10	—
2/XII	25°18′S 157°10′E	3 759	0	10±2	—	—
			500	10±2	15±3	1,5
			1000	12±3	12±6	1,0
			2000	5±2	10	—
12/XII	33°01′S 165°04′E	3 131	0	10±2	16±7	1,6
			500	5±2	—	—
			1000	5	—	—
			2000	5	—	—
16/XII	32°03′S 177°20′E	10 000	0	11±2	10	—
			200	7±2	10	—
			1000	5±2	10	—
16/XII	32°03′S 177°20′W	10 000	2000	5	10	—
			4700	5±2	10	—
			9000	5	10	—
18/XII	28°43′S 176°06′W	8 557	0	9±2	16±7	1,7
			500	5	10	—
			7000	5	10	—
19/XII	22°57′S 174°35′W	10 141	0	9±2	10	—
			500	7±3	—	—
			1000	6±2	10	—
			2000	5±2	10	—
			3000	5±2	10	—
			5200	5	10	—
			8000	5	10	—
21/XII	20°13′S 173°10′W	9 527	0	30±5	—	—
			500	16±3	—	—
			1000	5	10	—
			2000	5±2	—	—
			3000	5	10	—
			4000	5	10	—
			8000	5	10	—
26/XII	17°57′S 158°03 W	4213	0	26±4	26±9	1,0
			750	18±3	19±4	1,0
			2000	18±3	20±9	1,1
30/XII	18°43′S 144°54′W	4554	0	13±2	22±8	1,7
			500	10±2	15±8	1,5
31/XII	19°25′S 142°32′W	4418	0	10±2	17±7	1,7
			500	5	10	—
			1000	5	10	—
			2000	5	—	—
			3000	5	10	—
1967						
1/I	20°14′S 140°24′W	4585	0	12±2	18±9	1,5
			500	5	10	—
			1000	5	10	—
			2000	5	10	—
			4000	5	10	—
2/I	19°50′S 138°27′W	5057	0	13±2	15±8	1,1
			200	7±2	12±6	1,7
			500	7±2	—	—
			1000	5	10	—
			2000	—	—	—
4/I	16°05′S 137°36′W	4039	0	9±2	16±6	1,7
			500	5	—	—
			1000	5	10	—
			2000	5	10	—
			3000	5	10	—
5/I	11°40′S 135°43′W	4178	0	7±1	13±7	1,8
			500	5	10	—
			1000	5	10	—
			2000	5	10	—
			3000	5	10	—
8/I	1°15′S 134°59′W	4183	0	9±2	—	—
			500	5±2	—	—
12/I	13°14′N 131°04′W	5207	0	17±1	—	—
			500	5	10	—
			1000	5	—	—
			2000	5	10	—

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TABLE 2. (Continued)

Date of sampling	Coordinates	Depth, m	Depth at which the sample was taken	Concentration, decay/min · 100 liter Sr^{90}	Concentration, decay/min · 100 liter Cs^{137}	Cs^{137}/Sr^{90}
25/I	43°22′N	2938	0	126±18	100±12	0.8
			500	111±17	45±18	0.4
			1000	21±3	—	—
26/I	44°18′N 126°06′W	2941	0	106±15	119±12	1.1
			500	65±10	—	—
			1000	17±3	20±10	1.2
26/I	44°37′N 125°37′W	2921	0	90±13	123±14	1.3
			500	52±8	—	—
			1000	40±6	26±12	0.6
27/I	45°13′N 125°07′W	1607	0	65±10	110±12	1.6
			500	16±3	16±8	1.0
			1000	8±1	14±7	1.7
27/I	45°32′N 124°51′W	961	0	64±10	98±10	1.5
			500	6±2	10	—
27/I	46°14′N 124°43′W	750	0	60±9	89±10	1.5
			200	41±7	23±7	0.6
			500	30±5	18±9	0.6
28/I	46°35′N 124°35′W	803	0	55±8	76±10	1.4
			500	19±3	10	—
28/I	46°52′N 124°48′W	178	0	45±7	73±10	1.6
			100	35±5	65±10	1.8
29/I	46°32′N 126°28′W	2678	0	75±11	107±12	1.4
			500	5±1	—	—
			1000	5±1	10	—
5/II	22°13′N 110°16′W	3200	0	39±6	—	—
			500	5	—	—
			1000	5	—	—
			2000	5	—	—
Atlantic Ocean						
1967						
16/II	11°24′N 80°18′W	3500	0	9±1	—	—
			500	2±1	—	—
			1000	1±1	—	—
			2000	1±1	—	—
20/II	21°48′N 86°10′W	1480	0	11±2	—	—
			200	7±1	—	—
			500	10±2	—	—
			800	3±1	—	—
2/III	25°00′N 90°50′W	3500	0	9±1	—	—
			500	7±1	—	—
			1800	10±2	—	—
			2000	1±1	—	—
			3000	3±1	—	—
3/III	27°12′N 88°41′W	2500	0	3±1	—	—
			500	3±1	—	—
			1000	4±1	—	—
4/III	26°06′N 86°50′W	3100	0	3±1	—	—
			500	16±3	—	—
			1000	8±1	—	—
			1800	27±4	—	—
9/III	27°50′N 79°42′W	420	0	17±3	—	—
			100	5±1	—	—
			200	4±1	—	—
			300	2±1	—	—
10/III	28°04′N 79°31′W	730	0	22±4	—	—
			200	21±3	—	—
			500	16±3	—	—
			700	1±1	—	—
11/III	29°36′N 78°45′W	790	0	19±3	—	—
			200	7±1	—	—
12/III	29°58′N 77°55′W	800	0	15±2	—	—
			700	3±1	—	—
17/III	24°30′N 65°09′W	5500	0	5±1	—	—
			500	4±1	—	—
22/III	14°00′N 52°14′W	4800	0	4±1	—	—
			500	1±1	—	—
			1000	8±1	—	—
			4000	2±1	—	—
27/III	1°00′C 41°58′W	4300	0	11±2	—	—
			500	3±1	—	—
			1000	1±1	—	—
			2000	1±1	—	—
2/IV	0°20′S 37°28′W	4500	0	3±1	—	—
			500	1±1	—	—
			1000	1±1	—	—
			2000	3±1	—	—
			2800	3±1	—	—
3/IV	1°31′S 34°56′W	4000	0	21±3	—	—
			500	6±1	—	—
			1000	4±1	—	—
			2000	2±1	—	—
4/IV	0°27′N 34°25′W	4000	0	12±2	—	—
			500	5±1	—	—
			500	8±1	—	—
			1000	4±1	—	—
			3200	1±1	—	—
11/IV	25°19′S 41°45°W	3600	0	10±2	—	—
			500	36±6	—	—
			1000	4±1	—	—
29/IV	0°07′N 4°48 W	5170	0	5±1	—	—
			500	8±1	—	—
			2000	1±1	—	—
			3400	7±1	—	—
8/V	13°16′N 17°36′W	800	0	6±1	—	—
			200	3±1	—	—
			500	5±1	—	—
18/V	42°42′N 4°33′E	1200	0	8±1	—	—
			1	500	1±1	—
			1000	1±1	—	—
18/V	42°45′N 5°14′E	1200	0	3±1	—	—
			500	2±1	—	—
			1000	1±1	—	—

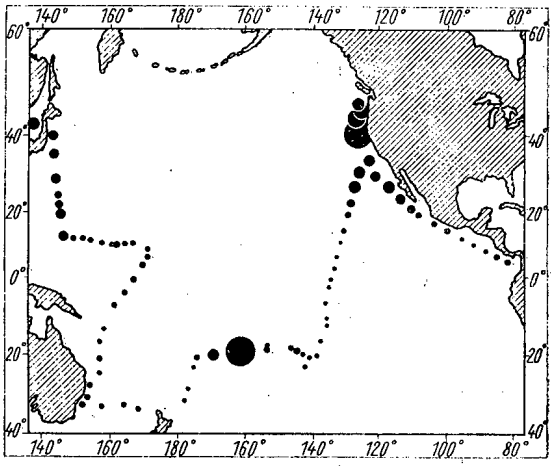

Fig. 1

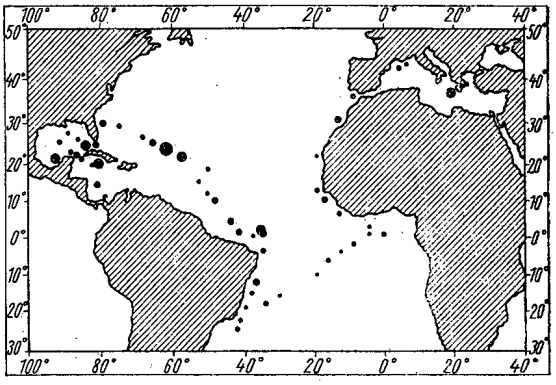

Fig. 2

Fig. 1. Sr^{90} concentration in the surface waters of the Pacific during 1966-1967 (the diameter of the dots is proportional to the concentration).

Fig. 2. Sr^{90} concentration in the surface waters of the Atlantic in 1967 (the diameter of the dots is proportional to the concentration).

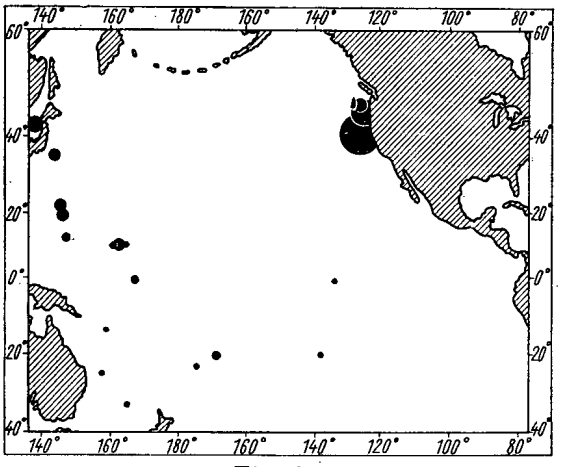

Fig. 3

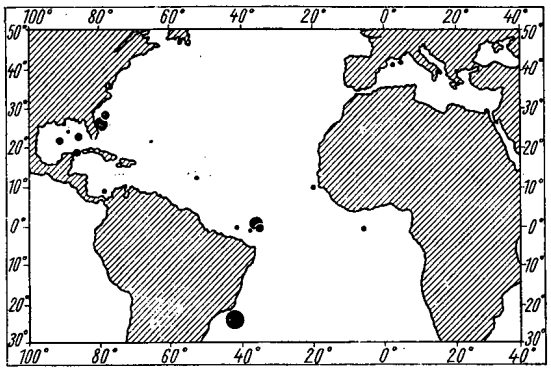

Fig. 4

Fig. 3. Sr^{90} accumulated in the water layer extending to a depth of 500 m per unit area of the Pacific.

Fig. 4. Sr^{90} accumulated in the water layer extending to a depth of 500 m per unit area of the Atlantic.

500 m, and to 135 and 162 mCi/km^2 down to a depth of 1000 m. According to the data which the Scientific Committee of the UNO on the Effect of Atomic Radiations obtained from continental stations, one had to expect 60-80 mCi/km^2 of Sr^{90} in this region [11]. Obviously, an excess amount of Sr^{90} is present.

The waters of the Northwestern part of the Pacific were contaminated by atomic weapons tests on the Marshall Islands. A region with extremely strong radioactive contamination was established in that area in 1954 and 1955. The situation was completely different in 1966. The comparison of the results with the data of previous years [12-15] is displayed in Fig. 5 which includes data referring to the Atlantic [5, 6]. The Sr^{90} concentration in the surface waters of this region decreased strongly during the last few years and amounted to an average of 30 decays/min · 100 liter water in the fall of 1966. The Cs^{137} concentration amounted, on the average, to 49 decays/min · 100 liter. The zone with increased concentration in the region of the Bikini–Eniwetok Islands had almost disappeared. The Sr^{90} and Cs^{137} amounts stored in the upper water layer extending to a depth of 500 m differed no longer from the concentrations in other parts of the Northwestern Pacific. However, in the upper, 1000 m thick layer, 115 mCi/km^2 of Cs^{137} are found above the Mariana trench. The concentration is below the detection limit at greater depths in the Mariana trench area. Samples have been taken to depths of 8000 m.

TABLE 3. Sr^{90} and Cs^{137} Concentrations at Various Points of the Water Layer Extending from the Surface to a Depth of 1000 m

Date of sampling	Coordinates at which sample was taken		Concentration (mCi/km²) in the layer extending from the surface to a depth of			
	latitude	longitude	500 m		1000 m	
Pacific Ocean						
1966						
1/XI	42°19′N	134°32′E	95	117	135	162
6/XI	36°00′N	142°41′E	70	—	95	—
10/XI	22°30′N	145°46′E	70	125	115	180
11/XI	19°36′N	147°25′E	73	—	98	—
13/XI	13°59′N	147°08′E	50	—	—	—
18/XI	11°42′N	161°30′E	43	—	63	—
19/XI	12°08′N	162°10′E	55	85	85	125
20/XI	12°05′N	164°46′E	26	50	46	80
25/XI	0°09′N	167°11′E	50	65	70	100
29/XI	13°26′S	158°42′E	15	—	30	—
2/XII	25°18′S	157°10′E	25	—	50	—
12/XII	33°01′S	165°04′E	20	—	—	—
16/XII	32°03′S	177°20′W	—	—	32	—
19/XII	22°57′S	174°35′W	20	—	35	—
21/XII	20°13′S	173°10′W	50	—	—	—
30/XII	18°43′S	144°54′W	30	45	—	—
1967						
2/I	19°50′S	138°27′W	19	—	—	—
8/I	01°12′S	134°59′W	15	—	—	—
25/I	43°22′N	127°58′W	270	165	420	—
26/I	44°18′N	126°06′W	195	—	290	320
26/I	44°37′N	125°37′W	160	—	270	340
27/I	45°13′N	125°07′W	90	145	120	180
27/I	45°32′N	124°51′W	80	—	—	—
27/I	46°14′N	124°43′W	94	77	—	—
28/I	46°35′N	124°35′W	85	—	—	—
29/I	46°41′N	125°45′W	90	125	110	—
29/I	46°32′N	126°28′W	90	—	100	—
Atlantic Ocean						
1967						
16/II	11°24′N	80°18′W	50	—	—	—
20/II	21°48′N	86°10′W	20	—	—	—
2/III	25°00′N	90°50′W	18	—	22	—
4/III	26°06′N	86°59′W	21	—	49	—
10/III	28°04′N	79°31′W	44	—	—	—
12/III	29°58′N	77°55′W	24	—	—	—
17/III	24°30′N	65°09′W	10	—	—	—
22/III	14°00′N	52°14′W	5,6	—	17	—
27/III	1°00′N	41°58′W	16	—	—	—
2/IV	0°20′S	37°28′W	4,5	—	—	—
3/IV	1°31′N	34°56′W	30	—	41	—
4/IV	0°27′S	34°25′W	21	—	27	—
11/IV	25°19′S	41°45 W	60	—	105	—
29/IV	0°07′N	4°48 W	15	—	—	—
8/V	13°16′N	17°36 W	10	—	—	—

TABLE 4. Comparison of the Average Sr^{90} and Cs^{137} Concentrations in the Northern and Southern Parts of the Pacific and Atlantic at Low Latitudes

Ocean	Radioactive isotope	Year	Concentration (decays/min · 100 liter)		Ratio North/South
			north	south	
Pacific	Sr^{90}	1961 [13, 14]	60	30	2,0
	Sr^{90}	1966	30	14	2,1
	Cs^{137}		49	24	2,0
Atlantic	Sr^{90}	1963 [6]	39	22	1,8
	Cs^{137}		77	35	2,2
	Sr^{90}	1964 [6]	29	18	1,6
	Cs^{137}		50	26	1,9

The Sr^{90} and Cs^{137} concentrations in the surface waters were lower in the Southwestern Pacific than in the Northwestern Pacific. Table 4 lists a comparison of the average concentrations and includes data referring to the Atlantic [6]. The ratio was maintained over the years. The Sr^{90} and Cs^{137} concentrations detected over the deep-sea Kermadek trough were maintained only in the upper water layer. The deep-sea waters at depths ranging to 9000 m were pure though according to preliminary hydrological data, these waters are not completely stagnant [16] and may be supplemented by sinking water masses from other areas.

The average Sr^{90} concentration amounted to 57 decays/min · 100 liter in the narrow latitudinal zone ranging from 17° S to 23° S. A point at which the Sr^{90} concentration reached 180 decays/min · 100 liter was determined from the contamination of the surface water in the region of the Tonga deep-sea trench. A high density of the radioactive fallout from the atmosphere was established from samples taken in this area. This is probably also the explanation for the increased Sr^{90} concentration at this point (18°47' S and 160°35' W). The Sr^{90} concentration was below the detection limit at all depths below the surface waters over the Tonga trench. The Sr^{90} concentration in the surface waters at other points of this area was in no way increased. However, according to the results of [17], Ce^{141}, Ce^{144}, Ru^{103}, Zr^{95}, and Nb^{95} were found in plankton samples taken at those points. The Sr^{90} and Cs^{137} concentrations were not high in the surface waters sampled between the Tuamotu archipelago and the coast of California. The Sr^{90} concentration on the Southern hemisphere was 17 decays/min · 100 liter on the average, and amounted to 35 decays/min · 100 liter in the Northern hemisphere between the equator and San Francisco. The Cs^{137} concentration reached average values of 18 decays/min · 100 liter in the area of Polynesia.

Increased Sr^{90} and Cs^{137} concentrations were found in January 1967 in Northern latitudes in the region where the Columbia river discharges its water. Our results should be compared with the data obtained by American scientists in 1964 [18]. In 1964, the average values amounted in this region (decays/min · 100 liter): in July 1964, Sr^{90} – 109; Cs^{137} – 102; Cs^{137}/Sr^{90} ratio – 0.9; in January 1967, Sr^{90} – 78; Cs^{137} – 100; Cs^{137}/Sr^{90} ratio – 1.3. A strong dilution by fresh water is typical for the sea water near the coast. In summer, the river waters were characterized by a Cs^{137}/Sr^{90} ratio of about 0.3. The ocean water was

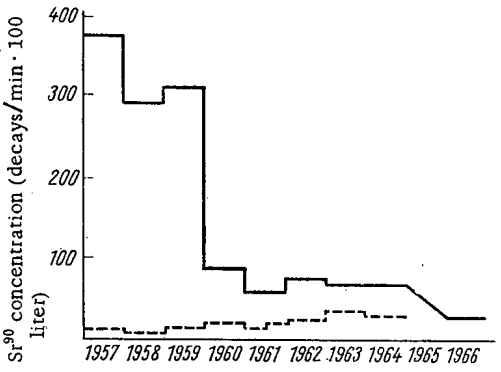

Fig. 5

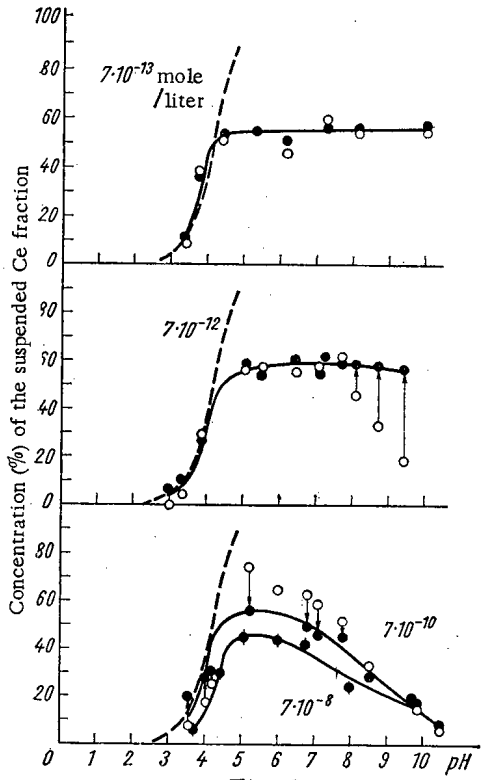

Fig. 6

Fig. 5. Change of the Sr^{90} concentration in the surface waters of the North Pacific and of the Atlantic during 1957-1966: ——) Pacific; – – –) Atlantic.

Fig. 6. Influence of the pH of the solution and of the total concentration of the element upon the formation of the suspended Ce^{144} fraction in double distilled water.

characterized by normal salinity (35%) and a Cs^{137}/Sr^{90} ratio of about 1.8. An intermediate value of 1.3 was observed in our work. This value was closer to the values obtained for the open sea than the river-water figures (70% of value for waters of the open sea, and 30% of value for river water); the salinity figures did not indicate this ratio. Thus, the radioactive contamination in 1967 was the same in that region as in 1964, though the admixed river-water quantities were reduced. The absolute values of the Sr^{90} and Cs^{137} concentration in the surface waters were there much higher than in other areas of the Pacific.

In this region, the Sr^{90} concentration was determined at six points, and the Cs^{137} concentration at three points (1000 m thick water layer in each case). The corresponding values (100-420 mCi/km^2 Sr^{90} and 180-340 mCi/km^2 Cs^{137}) are far beyond the figures which are characteristic for the rest of the Pacific. A comparison with data on the accumulation of fallout Sr^{90} [11] has revealed that the global radioactive fallout could provide only about 20% of the observed concentration. In an ocean area of about 220,000 km^2 (circle with a radius of 435 km) having its center in the mouth of the Columbia river, the Sr^{90} concentration was estimated at about 70,000 Ci and the Cs^{137} concentration at about 100,000 Ci.

The Sr^{90} concentration in the Pacific near the American coast between 35° N and 5° N was in 1967 about the same as in the Northwestern part of the Pacific, amounted, according to the average of 13 measurements, to 31 decays/min · 100 liter, and decreased from North to South.

Lists of Sr^{90} concentrations in the Atlantic were given in [19, 20]. However, for 1967 results of the measurements were stated only for medium latitudes. Our investigations in 1967 comprised mainly the tropical zone of the Atlantic.

The waters of the Caribbean are provided by the North equatorial current. The average Sr^{90} concentration amounted to 11 decays/min · 100 liter and did not differ from the Sr^{90} concentration in the tropical ocean area. The Sr^{90} concentration in the deep-sea water was below the detection limit. The Sr^{90} concentration in the Gulf of Yucatan was practically unchanged from the surface to a depth of 500 m.

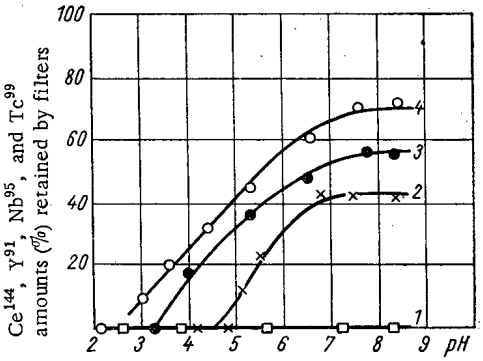

Fig. 7

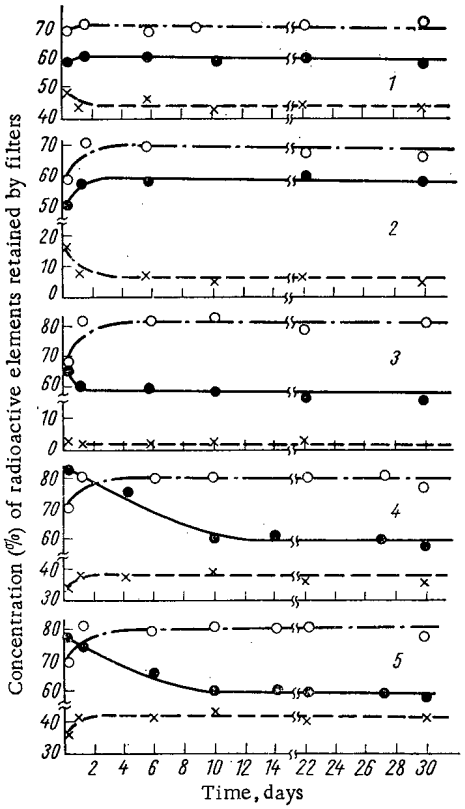

Fig. 8

Fig. 7. Amount of an element retained by an ultrafilter in dependence on the pH of sea water: 1) Tc^{99}; 2) Y^{91}; 3) Ce^{144}; 4) Nb^{95}.

Fig. 8. Formation of the suspended Ce^{144} fraction (———), Y^{91} fraction (– – –), and Nb^{95} fraction (– · – · –) as a function of time in waters taken from various depths: 1) surface; 2) 100 m; 3) 500 m; 4) 4000 m; 5) bottom water.

Artificial radioactive materials are probably discharged from the continent into the Gulf of Mexico. Diffusion and advective transfer cause a transformation of the water masses which are afterwards carried away by the Gulf Stream. An increased Sr^{90} concentration of up to 18 decays/min · 100 liter was observed in this area. The radioactive contamination reached to depths of 1800 m at one of the observation points.

Ocean waters from the Northern equatorial current circulate in the Western part of the Sargasso Sea. The Sr^{90} concentration amounted to 8-10 decays/min · 100 liter.

The average Sr^{90} concentration amounted to 11 decays/min · 100 liter in the equatorial zone of the Atlantic between 5° N and 5° S (spring of 1967). Unlike in 1963, no higher concentrations than in the neighboring areas were observed [6, 20]. The spot with an abnormally high concentration observed in the fall of 1963 had disappeared [6].

The Sr^{90} distribution had reached equilibrium in the southern part of the tropical zone by 1967. The average Sr^{90} concentration amounted then to 9 decays/min · 100 liter.

The average Sr^{90} concentration amounted to 10 decays/min · 100 liter in the zone ranging from 5° to 25° N.

The migration of Ce^{144}, Y^{91}, Nb^{95}, and Tc^{99} depends mainly upon the state of the radioactive isotopes. It is practically impossible to foresee which forms occur in the physical and chemical equilibrium in sea water. The specific adsorption laws, colloid formation, and other processes influence the behavior of microamounts of a substance in solution. The ocean medium, which contains dissolved and colloidal organic matter, bacteria, marine organisms, and admixtures of various compositions and origins, plays an important role.

TABLE 5. Retention of the Radioactive Elements under Consideration by Surface Suspension

No. of the suspension sample	Center of the region at which the sample was taken		Retention k_d of radioactive element			
	latitude	longitude	Y^{91}	Nb^{95}	Tc^{99}	Ce^{144}
1	22°42'S	170°34'3	$4{,}1 \cdot 10^3$	$5{,}8 \cdot 10^3$	0	$1{,}9 \cdot 10^3$
2	12°25'S	156°36'3	$4{,}3 \cdot 10^3$	$5{,}6 \cdot 10^3$	0	$1{,}9 \cdot 10^3$
3	21°06'N	173°04'3	$4{,}5 \cdot 10^3$	$6{,}1 \cdot 10^3$	0	$1{,}9 \cdot 10^3$
Station 5992	0°56'N	160°29'3	$4{,}3 \cdot 10^3$	$6{,}2 \cdot 10^3$	0	$1{,}7 \cdot 10^3$

Laboratory experiments in "pure" form were made with double-distilled water. The real behavior was studied in samples of sea water. Radioactive elements were introduced in the form of dissolved ions into the liquid phase.

The study of the state of the radioactive isotopes in the liquid phase involved ultrafiltration, centrifuging of solutions, ion exchange, and electrophoresis. The radioactive fraction which passed through a filter in ultrafiltration or which remained in the upper part of the test tube after centrifuging was assumed to be dissolved. The rest of the element was assumed to be in the "suspended" fraction. Extremely small amounts of the elements could be present in the solutions in the form of dispersed ions, molecules, genuine colloids, or pseudocolloids [21, 22].

Investigations of the Ce^{144} distribution in the dissolved and suspended fractions in double-distilled water were made in dependence on the pH of the solution and at concentrations ranging from $7 \cdot 10^{-13}$-$7 \cdot 10^{-10}$ M. These investigations revealed that in acid solutions at pH < 5, cerium remains almost completely in solution at extremely low concentrations (Fig. 6). According to [23], trivalent cerium cannot form genuine colloidal $Ce(OH)_3$ particles under these conditions at pH = 9.3. Colloidal $Ce(OH)_3$ particles could be retained on an ultrafilter. Tetravalent cerium which results in a colloidal fraction in the form of an hydroxide even at pH = 3.9 [24], is not formed under these conditions. If $Ce(OH)_4$ were present in the system, the colloid formation at high pH values would be more intensive. The dashed lines in Fig. 6 correspond to the theoretically calculated formation of trivalent cerium hydroxide [25]. It follows from Fig. 6 that the calculations agree with the experimental results only in the initial sections of the curves which represent the formation of the suspended fraction. This means that pseudocolloids are formed. Dissolved silicic acid [26] may play a role in the formation of these pseudocolloids. The reduced suspended fraction which is obtained when the concentration of the cerium ions is increased 100 times is probably given by ion-polymerization processes. Y^{91}, Nb^{95}, and Tc^{99} do not pass completely into the suspended fraction in double-distilled water, whatever the pH value [27-29].

In addition to the hydrolytical properties of an element, sorption processes occurring on suspended particles and the formation of complexes involving organic binding materials influence the properties of an element in sea water. The composition of both the suspended material and the organic matter are dissimilar at various points and at various depths in the ocean. This is the reason why an element can migrate in different ways during the various stages of its presence in the ocean. In order to check this assumption, the conversion of Ce^{144}, Y^{91}, Nb^{95}, and Tc^{99} ions introduced into water samples from various depths of the Pacific (at the point 00°56' N and 160°29' W) was studied [30, 31].

A check was made of the hypothesis that rare-earth elements precipitated with the sediments onto the ocean floor can be reversely reintroduced into the water layer by complex formation with organic, phosphorus-containing compounds [32]. The behavior of Ce^{144}, Y^{91}, Nb^{95}, and Tc^{99} in muddy waters obtained from the surface layer of ocean-bottom sediments at a depth of 5173 m at that point was studied.

Figure 7 displays some results of the experiments involving ultrafiltration. The formation of the suspended fraction is less intensive at increasing pH values than in double-distilled water. Technetium remained in the entire acidity range of the medium in its initial ionic state (ammonium pertechnate). As in the case of pure water, whatever the pH value, Ce^{144}, Y^{91}, and Nb^{95} are not completely converted into the suspended fraction. It was also shown that at pH > 7 and concentrations in excess of $7 \cdot 10^{-12}$ no reduction of the suspended Ce^{144} fraction is observed in sea water. This behavior seems to originate from the reduced polymerization of hydrolyzed ions by admixture of secondary materials in sea water. However, there may exist another reason for the observed behavior. According to [33], the oxidation of Ce^{3+} to the 4+ state is possible in sea water at pH > 7. A genuine colloid of pure dissolvable cerium compounds, e.g., cerium hydroxide or cerium phosphate, must therefore be formed.

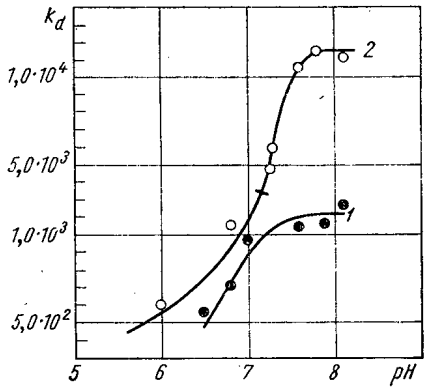

Fig. 9

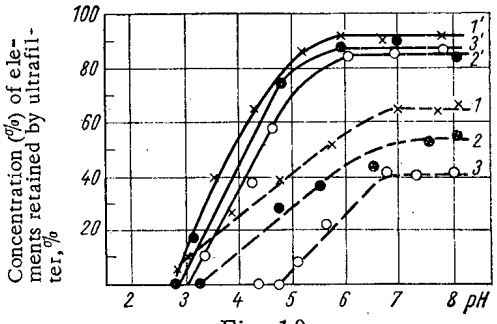

Fig. 10

Fig. 9. Equilibrium values of the distribution coefficients of Ce^{144} and Y^{91} in the sorption of these elements from sea water on anion-exchange resin Dowex-1: 1) Ce^{144}; 2) Y^{91}.

Fig. 10. Influence of iron upon the state of radioactive elements in sea water: – – –) natural sea water; ——) sea water at which 10 μg/liter Fe^{+3} had been added; ●) Ce; ○) Y; ×) Nb.

The dependence on the acidity of the sea water is the same for Y^{91} and Nb^{95} as for Ce^{144}.

Figure 8 displays the results of an investigation of the radioactive element distribution in waters taken from various depths. The Ce^{144} and Nb^{95} concentrations in the suspended fraction are practically the same in waters of all depths. The concentration of suspended Y^{91} is very low (0-10%) in waters taken from depths of 100 m to 500 m, but is increased (to 40-50%) in waters taken from the surface, from depths of 4000 m, and waters of deep-sea sediments. Obviously, Y^{91} interacts with certain complex-forming materials whose concentration increases from the surface to a certain depth but decreases thereafter. It is possible that these products stem from the decomposition of organic matter. The difference in the development of Ce^{144} and Nb^{95} on the one hand and Y^{91} on the other must imply a differentiation of these isotopes when they are transferred into the deep sea, and must also lead to an increased sedimentation of Ce^{144} and Nb^{95}.

Identical Ce^{144} and Y^{91} concentrations among the suspended and dissolved fractions were obtained by ultrafiltration and centrifuging. Centrifuging allowed calculations of the average radius of the colloidal particles, which amounted to 0.44-0.05 μ for Ce^{144} and 0.20-0.05 μ for Y^{91}.

The results obtained by ion exchange of solutions in double-distilled water supplement each other and agree with the theoretical conclusions concerning the interaction of pseudocolloidal particles with anion-exchange and cation-exchange resins [23, 24]. The sorption of the radioactive isotopes considered by the cation-exchange resin is practically zero at high pH values.

Cerium and yttrium of sea water are retained only by the anion-exchange resin (Fig. 9). At pH = 8.0, which is characteristic of natural sea water, 80% of the Ce^{144} and 30% of the Y^{91} in the solution are retained by the anion-exchange resin. The partial sorption of these elements by the anion-exchange resin and their absence in the cation-exchange resin probably originate from the formation of negatively charged particles in the hydrolysis process. The difference in the separation of these elements by ultrafiltration (or centrifuging) on the one hand and ion exchange on the other seems to be related to the formation (in addition to the negatively charged particles of colloidal aggregates) of aggregates consisting of the elements under consideration on the one hand and phosphates in sea water or dissolved complex compounds with organic substances on the other. The results agree with those obtained in electrophoretic studies, according to which negatively charged particles (which at pH < 7 had a positive charge) appear at pH > 7.5.

When the radioactive elements under consideration get into sea water, they undergo hydrolysis and negatively charged colloidal particles are formed (pH = 7.8-8.2). Apart from this, Y^{91} seems to form neutral charged particles [sic] with phosphates and dissolved complex compounds with organic binding substances.

Living matter which continuously removes chemical elements from sea water and introduces them into the biological cycle is one of the components of the sea-water suspension. Iron, whose concentration

in the suspended matter varies between 0.5% and 19.5% is another important component of the sea-water suspension [35, 36]. By forming iron hydroxide, iron can participate in the extraction of elements from sea water.

The influence of iron upon the state of Ce^{144}, Y^{91}, Nb^{95}, and Tc^{99} in sea water was therefore investigated with the ultrafiltration technique.

It follows from the results, which are displayed in Fig. 10, that almost 90% of Y^{91}, Ce^{144}, and Nb^{95} are converted to the suspended fraction when iron is added to sea water. About 45% of Y^{91}, 60% of Ce^{144}, and 70% of Nb^{95} are in the suspended state in natural sea water. The presence of iron does not affect the physical and chemical state of Tc^{99}. The results prove clearly that the removal of radioactive elements from the ocean implies certain dangers. The iron is transferred to plankton organisms via their food intake and hence, the iron per se is a carrier of fission fragments and favors the introduction of the radioactive isotopes under consideration into the biological cycle.

It was established that the coefficient of accumulation of all isotopes by plankton is in the presence of iron by an order of magnitude greater than in the sea-water–plankton–isotope system. An influence of the type of the plankton involved upon the results could not be established.

Apart from biogenic processes, which affect the distribution of the radioactive elements in sea water, nonbiogenic processes were also studied, i.e., the retention of isotopes by the surface of the suspension and by deep-sea sediments was considered.

Since the composition of the suspended matter is not a homogeneous function of depth, studies of the retention of radioactive isotopes by the suspension at various depths were of interest. Samples were taken from the depths 0, 100, 500, and 4000 m at the point mentioned above and also from the surface in various areas of the tropical Pacific. It was found that independent of the geographical location of the sampling point, the Ce^{144}, Y^{91}, Nb^{95}, and Tc^{99} quantities extracted by the suspension from the water are practically everywhere the same.

The investigations of the retention of the radioactive elements by the suspensions sampled at various depths have shown that the retention depends quantitatively upon the properties of the suspension and its granulometric composition. It was noticed in the case of the radioactive elements under investigation that the surface suspension is characterized by minimum retention and that the retention increases proportionally to the depth at which the sample was taken. The maximum retention was determined for a deep suspension which has a considerable amount of mineral grains in its composition. It was found that the retention of the radioactive isotopes was irreversible on those depth levels.

Secondary dissolution, mineralization of the suspensions, and biological processes contribute to the migration of the radioactive elements retained by the suspension. Due to the rapid sequence of plankton generations, the radioactive elements are retained again and again by plankton organisms and, hence, remain in the biological cycle in which they are transferred from the lowest organisms to higher organisms. Contrary to the idea of an unlimited dilution of the radioactive elements in the ocean, this transfer poses a direct danger to men.

In order to determine the influence of deep-sea sediments upon the distribution of the radioactive elements in the ocean, the most frequently occurring typical silts from the Indian and Pacific oceans were used as samples. The quantitative extraction of the radioactive isotopes from the sea water by ocean-bottom sediments was studied and it was found that Ce^{144}, Y^{91}, and Nb^{95} are almost completely retained by silts which contain more than 50% kaolite in their finely dispersed fractions. A slightly lower percentage of the isotopes (about 85%) is bound to the tropical radiolarian fine-aleuritic ooze which contains more than 16% of amorphous silica in the form of skeleton residues of plankton organisms, namely of Radiolaria and partial diatoms. The lowest degree of isotope extraction is typical for the low-lime argillaceous silts (80%). Sediments with a high concentration of calcium (about 60%) and volcanic sands (about 75%) have the lowest retention of the radioactive isotopes. The ocean-bottom sediments can be classified in the following fashion, as far as their retention of isotopes from the sea water is concerned: carbonate-free argillaceous silts > tropical radiolarian fine aleuritic silts > low-lime argillaceous silts > carbonate silts. The data prove unambiguously that the degree of retention by the ocean-bottom sediments depends directly upon their surface area and the chemical composition of the sediments. For example, one of the good isotope-concentrating substances, namely fine aleuritic silt, has a very large surface (32.5 m^2/g) and contains 16.3% amorphous silica and 1% calcium carbonate, whereas a carbonate silt which is one of the poor absorbers has

a total surface of 0.61 m^2/g, is free of amorphous silica and consists almost entirely of calcium carbonate (93.5%).

The extraction of radioactive elements from sea water by ocean-bottom sediments is a complicated physical and chemical process which comprises the mechanical retention of colloid particles of the elements by silts and the sorption of the radioactive isotope fraction which is present in the sea water in genuinely dissolved form.

The results presented in this article and the data of [5-7] provide the overall picture of the radioactive contamination of the world oceans during 1966-1967. The initial conditions for further studies of the radioactivity in a certain ocean area are stated for each region which was examined for the first time.

The radioactive overall contamination of the surface waters had decreased by 1966-1967 and several previously detected radioactive spots had become more diffuse. The Sr^{90} amounts which had been transferred to the depths of the Pacific exceeded the amount precipitated by global radioactive fallout. Similar reliable data were obtained only at a few points in the Atlantic.

Radioactive regions, which are produced in international waters by modern technology, the steady increase in the number of these spots, the intensity of the contamination, and their area, as well as merging of such spots, are alarming circumstances.

In order to obtain a better understanding of what might happen with the ocean and its resources in the future, the physicochemical and biological processes controlling the removal of radioactive isotopes discharged into the ocean have been studied. Hydrolysis, complex formation, and concentrating in sea-water suspensions play an important role in the case of the solvable forms of several radioactive elements. The introduction of artificial radioactive isotopes into the biological cycle may result in an accumulation of these isotopes in certain sections of the migration chain. Carbonate-free argillaceous silts are the best absorbers of the radioactive isotopes considered among the various forms of ocean-bottom sediments. The mechanism of the extraction of these isotopes from sea water is a combined mechanical retention of colloidal particles of these substances by ocean-bottom sediments and the sorption of the radioactive fraction which is present in dissolved form in the sea water.

It follows from these considerations that the discharge of radioactive wastes into the ocean poses a certain risk. In view of the rapid development of the nuclear industry and energy production, one must already at the present time provide for efficient measures in order to avert the future radioactive contamination of the oceans and seas.

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POSSIBILITIES OF EXTENSIVE PEACEFUL USE OF ATOMIC ENERGY WHILE PROVIDING RADIATION SAFETY FOR THE POPULATION

Yu. A. Izrael' and E. N. Teverovskii

At the present time nuclear power engineering is based at atomic power plants (APP), which employ the fission of U^{235} with thermal neutrons [1]. Its future development is linked with fast neutron reactors using U^{238}. It is expected that the overall use of energy at the year 2000 will constitute $2.1 \cdot 10^{10}$ tons of conventional fuel. The power of an electric power plant will increase to $1.1 \cdot 10^7$ W, and the portion of the APP will rise to 50% ($4.3\text{-}5.5 \cdot 10^6$ MW) [2, 3]. The electrical efficiency of an APP, which is 25-30%, will rise to 42-45% [2]. Moreover, in the near future the development of new uses of atomic energy in industry and construction should be expected.

It is extremely important that the development of nuclear power engineering does not lead to harmful or dangerous consequences as a result of possible emanation of a certain quantity of radioactive materials into the external environment. This factor may exert a significant influence (and perhaps a decisive one) on the development of nuclear power engineering. Although there is not one branch of industry which does not have such means of protection against possible contamination of the external environment, as in nuclear power engineering [3], the question of its effect on the external environment is receiving increasing attention [3, 4]. Increasingly more effective means of protection are applied, estimates are made of the doses of possible emissions in the immediate vicinity of the APP, and the harm from the emissions of an APP is compared with that of enterprises operating on conventional fuel. At the same time, the natural radioactivity from emission from such enterprises is taken into account [3]. In certain cases the dose from natural radioactivity in operations using conventional fuel may even exceed the dose from radioactive products emitted by an APP [5].

However, for an evaluation of the prospects for the development of nuclear power engineering, estimates are needed on a global scale, i.e., estimates over the extensive areas encompassing the country and the continent. It is essential to have a prognosis of the global radiation situation, taking into account all probable emissions in the realization of possible development programs. It is necessary to determine the absolute quantity of radioactive products in the natural environment, the doses from these products, and to compare the possible consequences of contamination of the natural environment with that from nonradioactive harmful materials.

Damage through the use of conventional fuel is linked mainly with the emission of hamful materials which form during the combustion of the fuel in the atmosphere. In the operation of electric power plants and engines which use coal, black oil, and other products from the processing of petroleum, these are mainly sulfur dioxide and ash, hydrocarbons, nitric oxides, oxides of lead, etc.

For some time power installations throughout the world have been emitting 200-250 million tons of fuel ash in various soots and nearly 60 million tons of sulfur dioxide annually. The content of toxic materials in the air of a number of large cities significantly exceeds the permissible values [6].

At present the emission of sulfur dioxide in the burning of coal is 160-2040 g for one ton of the burnt fuel. The quantity of ash (taking into account ash trapping) may exceed the indicated values by several times [6]. According to the data in [6], the approximate formula for determining the quantity of sulfurous gas a, which is emitted by an electric power plant into the atmosphere in the burning of black oil, has the

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TABLE 1. The Relation of Calculated Concentrations q to the Maximum Permissible Concentrations in the Operation of an APP and an Electric Power Plant on Solid or Liquid Fuels

q:MPC (maximum permissible concentrations)	APP					Standard electric power plant	
	routine operation		accident situation				
	H^3	Kr^{85}	I^{131}	Sr^{90}	Cs^{137}	SO_2	volatile ash
On the surface of the earth . . . . . In the air.	$5 \cdot 10^{-6}$	$3 \cdot 10^{-3}$	$7 \cdot 10^{-4}$	10^{-6}	10^{-5}	10	10
Maximum amount for the given form of power engineering. .	$3 \cdot 10^{-3}$		$\sim 10^{-3}$			10	

Note: The quantities q:MPC are obtained for the assumed values for the altitude of the mixing layer. In changing the altitude of this layer these values are changed; however their relations to each other (for standard electric power plants and APP's) do not change.

form

$$a \sim 0.09 \rho \alpha,$$

where a is expressed in tons/day; ρ is the power of the plant in MW; α is the sulfur content of the black oil by %.

If we assume a mean value of $\alpha = 2.5\%$, then at the present time the annual emission of sulfur dioxide by thermal electric power plants (assuming an identical emission per unit of power during operation, using liquid and solid fuel, at overall powers of $\sim 10^6$ MW) is determined to be equal to approximately 80 million tons, and by the year 2000 it will consist of nearly 450 million tons of SO_2 and nearly 1.5 billion tons of ash (at powers of $5.5 \cdot 10^6$ MW [2]). Of course, the quantity may be reduced somewhat due to the application of highly efficient but costly means of purification.

In the operation of an APP, radioactive fission products gradually accumulate in the active zone of the reactor. The construction of modern plants practically eliminates the entrance of such products into the surrounding environment in their normal operation [4] (the problem of radioactive wastes is not considered in this paper). At the same time only an insignificant quantity of gaseous isotopes of induced activity – Ar^{41}, C^{14}, and H^3 – may enter the atmosphere. Thus, according to the data of [5], up to 650 Ci/day of Ar^{41} are emitted.

However, according to the data of [7], the radiation dose even in the vicinity of an APP does not exceed 1% of the allowed dosage. Tritium is formed in reactors as a result of fission and neutron capture with nuclei of deuterium and Li^6 (approximately in equal quantities), and up to 10-30 Ci/MW·year enters the atmosphere [8].

It is necessary to note that in the dissolution of spent fuel elements in radiochemical production the discharge of Kr^{85} (a naturally long-lived inert gas) into the atmosphere is evidently unavoidable. In this instance calculations must take into account that in the total fission of one ton of U^{235} (in approximately 800 MW·year of produced electric energy for an APP) $4.2 \cdot 10^5$ Ci of Kr^{85} are formed.

In an accident situation at an APP, emission into the atmosphere of isotopes of iodine and other relatively volatile isotopes is possible. The greatest emission in the whole history of reactor building (not less than 10^3 reactor·years [3]) took place during an accident at Windscale, which was accompanied by a fire in the active zone: $2 \cdot 10^4$ Ci of I^{131}; $1.2 \cdot 10^4$ Ci of Te^{132}; $6 \cdot 10^2$ Ci of Cs^{137}; 80 Ci of Sr^{89}; 2 Ci of Sr^{90} [9]. According to estimates in [10] the frequency of accidents at reactors is less than 10^{-2} for one reactor·year. The probability of large emissions of I^{131} is extremely small. In [11] it is assumed that a major accident (with an emission of several thousand Ci of I^{131}) may occur once in a thousand years of reactor operation. A curve estimating the possibility of emanation of I^{131} (the dependence of the emission of I^{131} on the number of years of reactor operation), is described by the law $t^{0.6}$. The degree of danger for incidence of cancer of the thyroid gland for regions with a high population density (up to 5000 people per km^2) is maximal for emanations of 10^3 Ci of I^{131} and consists of three cases [12].

For an estimate of the global radiation situation due to the development of nuclear power engineering we will proceed from the following assumptions: 1) by the year 2000 nearly $5 \cdot 10^3$ reactors with an average

power of 10^3 MW each will be put into operation; 2) the distribution of reactors by land is more or less uniform; 3) the number of significant accidents with a discharge of nearly 10^3 Ci of I^{131} is equal approximately to five (one accident for 10^3 reactor·years) or smaller accidents with a discharge of 10^2 Ci of I^{131} – nearly 30 per year. Thus in a year there will be emitted into the atmosphere, as a function of reactor operation time, up to $5 \cdot 10^3$ Ci of I^{131}, 30-140 Ci of Cs^{137}, 4-7 Ci of Sr^{89}, 0.1-1.0 Ci of Sr^{90} [8]. The isotopes indicated above will, in the course of several days, precipitate out onto the surface of dry land (taking into account the rate of effective precipitation equal to 0.1-1 cm/sec), without succeeding in diffusing throughout the entire globe. In calculations it may also be assumed that all these isotopes will precipitate out either on dry land areas or in the danger zone of reactors (nearly 10^7 km^2, neglecting possible overlap).

The most dangerous isotope in an accident (because of intake through inhalation and with milk) is I^{131}; the remaining isotopes present a danger mainly because of external radiation (Cs^{137}) or in entering into the organism with water and food (Sr^{89}, Sr^{90}). In agreement with our calculations and with the data of [8], contaminations of the surface of the earth leading to a critical radiation dose for the population (for a year) is 1.0-2.0 Ci/km^2 for Cs^{137}, 0.06-0.6 Ci/km^2 for Sr^{90}, and for I^{131} 50 mCi/km^2 (for children).

In the dissolution of fuel elements nearly $3 \cdot 10^9$ Ci for Kr^{85} per year may enter the atmosphere (according to the operating level of an APP until the year 2000). Long-lived isotopes of tritium and Kr^{85} will be uniformly mixed in the lower layer of the atmosphere of the whole earth. The dosage from tritium is significantly lower than from Kr^{85} and by the year 2000 will average 10^{-3} mR per year for the whole population [13].

The data averaged over the local affected areas (10^7 km^2) for various isotopes (the most dangerous) are presented in Table 1.

We assumed in these calculations that the diffusion (mixing) of radioactive isotopes, for which concentration in the air is critical, occurs in the lower 200 m layer of the atmosphere, and for Kr^{85} and H^3, in the 500 m mixing layer (in view of their duration in the atmosphere). In calculations of the contamination by conventional toxic materials we assumed uniform mixing also in the 200 m lower layer of the atmosphere. Taking the life time of SO_2 to equal four days [15] and the expected quantity of its emission into the atmosphere by the year 2000 to be $4.5 \cdot 10^8$ tons per year, we have found that in the air above dry land there will occur on the average a constant 10-fold excess of daily mean maximum permissible concentration (MPC) equal to 0.05 mg/m^3. Similar figures (relative to the MPC) are obtained with a calculation of the concentrations of volatile ash and other toxic materials. For radioactive materials the mean annual concentration does not exceed 10^{-3} of the permissible concentration.

The data presented in Table 1 show that for the above-indicated energy production, at conventional enterprises, the dilution of toxic materials up to the permissible level requires a 10^3-10^4 times greater quantity of air than in the operation of an APP. This relation does not depend on the assumed altitude of the mixing layer, unlike the values for the excess of the permissible concentration. Although the maximum contamination levels for radioactive and nonradioactive toxic materials are based on several different principles, it must be kept in mind that emissions of toxic materials (both radioactive and nonradioactive) may lead both to an immediate harmful effect on the population and to definite genetic effects [16].

We also note that the use of atomic energy does not require the consumption of oxygen necessary for the combustion of conventional fuel and does not lead to the continuous growth of the content of carbon dioxide in the atmosphere.

Thus it is possible to conclude that the development of atomic power engineering guarantees the preservation of sufficient purity of the external environment, and, moreover, the replacement of energy generated with conventional fuel by atomic energy leads to a significant decrease in the contamination of the external environment by toxic materials and the improvement of the sanitary conditions of the inhabited environment.

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THE USE OF ACCELERATORS IN MEDICINE AND THE NATIONAL ECONOMY

E. G. Komar

The use of accelerators of charged particles in medicine is extremely effective in conjunction with other methods of therapy. The use of accelerators for practical purposes in industry and agriculture permits a number of new technological processes to be carried out and substantially reduces the cost of production. Moreover, expenditures for the installation of accelerators usually pay for themselves during the first year of operation. Despite this, accelerators are still used to a limited extent. Three reasons for the slow development of accelerator radiation technology might be mentioned.

First of all, reliable, cheap, and convenient-to-operate accelerators have appeared only in recent years. At the present time, the industry of many countries, including the USSR, is already producing accelerators in sufficient numbers and assortment to fully satisfy the growing demand.

The second and evidently most important reason is the negligible number of technological processes developed.

And finally, the third reason is associated with the insufficient knowledge of specialists and is of a purely psychological nature. This is the fear of radiation processes, the lack of confidence of their effectiveness and safety. However, the statistics of recent years shows that the number of accelerators being put into operation is increasing. This is understandable, since:

1) accelerators can be shut off at any moment, after which they are safe, and they permit examination, repair, and other work;

2) the energy, intensity, and type of radiation can be varied within broad limits, according to a present program;

3) the radiation flux can be directed with the aid of electromagnetic lenses, scanning devices, etc;

4) any beams, as powerful as desired, necessary for the production of high output of processes, can be produced with accelerators;

5) accelerators have a high coefficient of utilization of the beam, of the order of 70%.

The irradiation output can be computed according to the following well-known formula:

$$M = \frac{\varepsilon P}{3R},$$

where M is the output of the installation, tons/h; ε is the coefficient of utilization of the beam; P is the power of the beam, kW; R is the dose received by the product, Mrad. Thus, for example, an accelerator with a power of 10 kW at a dose of 20 Mrad and $\varepsilon = 0.75$ can irradiate 125 kg of material per hour. The disinfestation of grain requires a dose 1000 times smaller (20 krad). In this case, an accelerator with a power of 10 kW can irradiate 125 tons of grain per hour.

The shortcomings of electronic accelerators include the comparatively low depth of penetration of electrons into the material to be irradiated. As is well known, the depth of penetration is

$$l = \frac{0.35W}{d},$$

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TABLE 1. Basic Medico-Technical Characteristics of Linear Accelerators and Betatrons

Accelerator	Energy of electrons, MeV	Energy of bremsstrahlung quanta, MeV	Dose rate of ionizing radiation at a distance of 1 m, R/min		Maximum dimensions of irradiation field, cm^2
			for electrons	for bremsstrahlung	
LUÉ-5	5	5	—	300	18×18
LUÉ-15	15	15	—	300	30×30
LUÉ-25	10—30	10, 15 20, 25	50—500	50—500	20×20
B5M-25	10—23	7—25	300	40	18×18
B1M-45	10—45	10—42	500	125	20×20

where W is the energy of the electrons, MeV; d is the density of the substance, g/cm^3. When W = 10 MeV, d = 1 g/cm^3, l = 3.5 cm. For most processes, the indicated depth is sufficient. Many processes require surface irradiation. In a number of cases the thickness of the material is small (film, ribbon), and therefore the use of low-energy accelerators, up to 1 MeV, is possible. When it is necessary to irradiate materials of greater thickness, the electron radiation can be converted to powerful bremsstrahlung.

Electron Accelerators for Medicine

At the present time, three models of linear electron accelerators (LUÉ-5, LUÉ-15, and LUÉ-25) and two models of betatrons (B5M-25 and B1M-45) are being developed, manufactured, and are in operation in medical institutions in the Soviet Union. The basic medico-technical characteristics of these accelerators are cited in Table 1.

The LUÉ-5 accelerator permits static and rotation irradiation to be conducted. It operates in a system of bremsstrahlung and has an outlet of electrons for research work. Multiyear operation of the accelerator under clinical conditions has demonstrated the high reliability of operation of the equipment. Usually up to 50 sessions of irradiation are conducted per shift.

The LUÉ-25 accelerator is a universal medical apparatus, providing precision radiotherapy with bremsstrahlung and electrons. The formation of the radiation fields of irradiation is produced by two therapeutic heads, permitting the creation of a bremsstrahlung field of any configuration and the production of a field of electrons without admixture of bremsstrahlung by the method of static electromagnetic scattering.

The development of a linear electron accelerator (LUÉ-15) for medical purposes, which will permit static tangential, and rotational–tangential irradiation, is now being completed. The dose rate is varied according to a set program, depending on the position of the radiation source. The small size of the electron spot on the target, as well as the precision collimation of the bremsstrahlung beam, permit the production of the uniform radiation field of any configuration with sharply outlined boundaries. The accelerator provides for systems of automatic introduction into the system, and detection of breakdowns.

The type of B5M-25 betatron is series-produced by industry; the type B1M-45 betatron is in the development stage. This machine permits static, rotational, and tangential irradiation. Both machines have remote control by collimating diaphragms and light indication of the dose field.

Accelerators for Practical Purposes

General Characteristics of Accelerators. Table 2 presents the basic characteristics of several accelerators manufactured in the USSR and used for flaw detection, activation analysis, the creation of new technological processes, increasing the productivity of agriculture, as well as for other practical purposes.

The technological processes, permitting a large dose rate, can be provided for by accelerators of the Élite (electronic pulsed transformer) type. However, in a number of cases the pulsed and average powers of the beam may not differ significantly. Certain technological processes do not permit large dose rates. In these cases it is necessary to select accelerators of the RTÉ (resonance transformer – accelerator of electrons), Élektron (cascade amplifier of voltage with inductive feedback), and KG (cascade generator) types. An accelerator of the Élektron type is supplied with its own biological shield and can be set up in any room.

Flaw Detection. Flaw detection of castings, rollings, welded seams, and other elements of objects, the strength of which should be guaranteed, can be carried out with the accelerators cited in Table 2 under the Nos. 1, 3, 4, and 10-14. The basic indices of these accelerators are the dose rate of x-radiation and the cross-sectional dimensions of the beam on the target. The first number determines the time of exposure, the second, the sensitivity of the method. Thus, for example, for an RTD-1 accelerator, at a dose

TABLE 2. General Characteristics of Accelerators for Practical Purposes

No. in order	Name of accelerator	Type of accelerator	Energy of accelerated electrons, MeV	Average power in beam, kW	Pulsed power in beam, kW	Yield of x radiation at a distance of 1 m from the target, R/min	Diameter of beam of target, mm	Size of accelerator, m		Weight of accelerator, tons
								diameter	length	
1	RTD-1	Resonance transformer	1,0	3	18	60	0,25	0,9	1,5	0,9
2	Élektron-1	Transformer	0,7	7	7	—	—	0,7	3,0	1,0
3	Élite-500	The same	0,5	1	700	18	—	0,3	0,5	0,04
4	Élite-1	" "	1,0	8	10 000	360	—	0,4	0,6	0,12
5	Élite-3	" "	2,5	10	40 000	—	—	1	1,3	0,8
6	ÉLT-2	" "	1,5	25	215	—	—	1,3	2,4	7,0
7	KGÉ-2.5	Cascade generator	2,5	20	20	—	—	3,0	6,2	32,0
8	LUÉ-8-5V	Linear accelerator	8,0	5	3 500	—	—	5,0· 0,7·1,75		2,0
9	LUÉ-13-9	The same	13,0	9	11 000	—	—	5,5· 1,5·2,35		5,0
10	LUÉ-10-1	Linear accelerator	10,0	1	1 000	2 000	1,5	2,75·1,0·0,8		2,0
11	LUÉ-15-1.5	The same	15,0	1,5	1 500	10 000	2,0	4,5·1,5·2,0		5,5
12	B-25	Betatron	25,0	—	—	40	—	—		2,5
13	B-35	The same	35,00	—	—	250	—	—		5,0
14	B-50	" "	50,0	—	—	800	—	—		20,0

rate of 60 R/min and cross-sectional dimensions of the beam 0.25 mm, the sensitivity during radioscopy is better than 2% of the thickness. This means that at a thickness of steel of 150 mm, inhomogeneity of the structure of the metal with dimensions 3 mm can be detected.

The most powerful flaw detectors of the machines indicated in Table 2 are accelerators Nos. 10 and 11, in which the dose rate is equal to 2000 and 10,000 R/min, respectively, at a distance of 1 m from the target. These accelerators permit radioscopy of large objects at high sensitivity within a few seconds.

Activation Analysis. Activation analysis is one of the effective regions of application of the accelerator technique. It is used in metallurgy and geological prospecting for rapid and accurate analyses of the content of elements. There are transportable and stationary activation analysis laboratories.

Transportable laboratories are used for work under field conditions. In the Soviet Union, a generator of the NGI type, with vacuum exhaust tube, is manufactured for these conditions in two variations: with a grounded target and with a target under voltage. The generators operate in a pulsed system with pulse duration 1-2 μsec and a frequency of 10-30 Hz. The neutron flux density directly around the target is of the order of 10^7 neutrons/cm^2, which is quite sufficient for work under weld conditions. The neutron energy is 14 MeV. The sensitivity of the generator is 0.5% in the case of the determination of aluminum in ores, 0.05% for copper, 1.5% for titanium, and 1% for manganese. The sensitivity in the determination of the oxygen content in copper is $1 \cdot 10^{-3}$%. Neutron generators of the NG-150 type are distributed for stationary activation analysis laboratories. The basic data on this generator are cited below:

Accelerator voltage	150 kV
Current of ion beam on the target	3 mA
Diameter of beam on the target	5-25 mm
Neutron flux in the D–T reaction	$2 \cdot 10^{11}$ neutrons/sec
Consumable power	8 kV-A
Water consumption	350 liters/h
Weight of accelerator	0.5 tons
Total weight of neutron generator	1.3 tons

The stationary laboratory based on this generator permits the analysis of the oxygen and nitrogen content with a sensitivity to $1 \cdot 10^{-5}$%.

The compact cyclotron now being developed in the USSR can be used as a universal source of various types for activation analysis. The basic parameters of the cyclotron are cited below:

Energy of helium-3 ions	8-26 MeV
Energy of protons	5-18 MeV
Energy of deuterium ions	3-10 MeV

The cyclotron has a high intensity of the inner and outer exit beam of up to 1 mA and 50 μA, respectively.

Certain Radiation Technological Processes, Developed and Used in the USSR. Let us enumerate briefly some of the radiation processes that have now been developed, are being used, or are beginning to be used on the basis of the accelerators cited in Table 2. Here we shall not mention the processes that are now in the stage of research and development.

1. As is well known, the treatment of polyethylene at a dose of 10-30 Mrad increases the thermal stability of the material to 250-300°C with conservation (and even improvement) of its mechanical and electrical properties. This process is used in the production of thermally stable cables and film, used by various branches of industry and agriculture. The energy of the accelerator is selected, considering the thickness of the material.

2. The preparation of linear accelerators of the LUÉ-8-5V type (see No. 8, Table 2) for the sterilization of products is being completed. The sterilization of surgical instruments and other medical equipment is proposed. The irradiation dose in sterilization is 2.5-3 Mrad.

3. The use of a radiation chemical process for the production of glass plastic is being suggested. This process will simplify the technology and improve the quality of the product. The labor efficiency is increased by 2.5-fold in comparison with the chemical method of production. The necessary irradiation dose is 7 Mrad.

4. Radiation treatment of rubber during its vulcanization ensured high resistance of the product to aging and wear. As the investigations indicated, for individual varieties of rubber this stability is increased by 20-30%.

5. The technology has been developed and the production of a self-adhesive insulation tape of polyheterosiloxane, 0.2-0.5 mm thick, has been begun. The new tape will replace the traditional mica tape in electrical machine production. After electron irradiation at an energy of 500 keV with a dose of 5 Mrad, the tape becomes sticky. After two days, insulation made out of this tape becomes monolithic without any additional treatment. Such insulation can operate for long periods at a temperature of 300°C and briefly at 600°C. The cost of the new insulation is half that of the usual insulation, made from mica. A 500 keV cascade generator, the data for which are not cited in Table 2, is used as the accelerator in the production of the self-adhesive tape.

The Use of Accelerators in Agriculture

As an example, we shall discuss here only one line – the electron irradiation of potatoes before planting to stimulate germination and before loading in the warehouse to suppress germination.

Irradiation of Potatoes. For seven years laboratory, greenhouse, field, and farm experiments on the irradiation of potato tubers with electrons have been under way in the USSR. The optimum energy of irradiation has been determined: 0.7-1.0 MeV. At this energy, the electrons penetrate to a depth of 3-4 mm into the tuber. The dose absorbed by the inner layers of the tuber and associated with bremsstrahlung does not exceed 0.1% of the dose received by the surface layers. Thus, in electron irradiation, the mass of the potato consumed for food is practically not irradiated.

Irradiation can be used to achieve two purposes: 1) irradiation of seed potato before planting to increase the yield; 2) irradiation of potato tubers before loading in the warehouse to prevent germination and spoilage.

At low doses of irradiation (75-300 rad), there is a stimulation of growth processes, leading to an increase in the yield from 15 to 30% or more, as well as to an increase of 1.5-2% in the content of dry substances, 1-1.5% of starch, and 4-5 mg % of vitamin C. In the case of irradiation with higher doses (10-20 krad), as a result of the suppression of growth processes, the periods of storage are substantially increased (practically to a new level) without any deterioration of the qualities of the product. In the case of irradiation of tubers in October with a dose of 20 krad and storage until September of the following year, the starch content is reduced by only 2.5%. In control, nonirradiated tubers this content is reduced by 24.9%. The vitamin C content in the irradiated tubers is also greater than in the nonirradiated tubers. It is suggested that the RTD-1 (see No. 1, Table 2) be used as the accelerator for the irradiation of potatoes in the mass application of this process.

Now a method has been developed for the treatment of patients with accelerators, and requirements have been formulated for the corresponding complex accelerator installation.

Many technological processes using fluxes of charged particles have already been developed, and the requirements for the corresponding apparatus have been formulated. Research work on the creation of new processes and new directions of the use of accelerators is continuing. Therefore, we should expect that in the immediate future the use of accelerators in medicine, industry, and agriculture will develop at a rapid rate.

INTERNATIONAL COOPERATION OF THE USSR IN THE FIELD OF THE PEACEFUL USES OF ATOMIC ENERGY

G. S. Afonin, Yu. G. Balasanov,
I. D. Morokhov, B. A. Semenov,
and I. I. Smolin

The seven-year period that separates us from the Third International Conference on the Peaceful Uses of Atomic Energy has been marked by the further development of atomic science and technology and the expansion of international cooperation.

During this period the Soviet Union has invariably presented proposals directed toward the alleviation of international stress, the strengthening of peace and cooperation, and the guarantee of the use of atomic energy for the welfare of progress and constructive development.

An extremely important practical step in this direction was the Moscow Treaty on the prohibition of nuclear weapons testing in three media (1963), which eliminated the most hazardous source of artificial radiation, capable of damaging the health of humans and doing irreparable harm to man's environment, as well as the Nuclear Arms Limitation Treaty (1970), which, reducing the danger of the unleashing of nuclear warfare, is opening up new prospects for the peaceful use of atomic energy, with the proper consideration of the needs of the developing regions of the world. Chairman of the Council of Ministers of the USSR, A. N. Kosygin, noted in his speech in the ceremony of signing of the agreement that participation of a wide circle of states in the signing of the treaty convincingly indicates that mutually acceptable solutions to complex international problems, vitally important to all mankind, can be found by the states.

At the present time, science has achieved a level of development such that many scientific problems can be solved only through creative cooperation, unification, and coordination of efforts at the international level.

The Soviet Union has unswervingly spoken out for such international cooperation in the field of the use of atomic energy, which would fully correspond to the aims and principles of the United Nations Charter, the International Atomic Energy Agency (IAEA), and the premises of the Nuclear Arms Limitation Treaty, since only on the basis of equality and a consideration of the interests of all countries is fruitful cooperation, promoting economic and social progress of all mankind, possible.

The Soviet Union is participating in scientific and technical cooperation with the socialist, developing, and industrially developed countries, as well as with international organizations. Each of these directions includes various forms of cooperation: conducting joint scientific research and experimental work with specialists from socialist and capitalistic countries, on the basis of bilateral and multilateral agreements; the rendering of technical aid and assistance in the creation of national atomic centers in socialistic and developing countries, the participation of Soviet scientists and specialists in international conferences, symposia, meetings of experts, and seminars; conducting mutual acquaintance visits on concrete topics of atomic science and technology; international exchange of scientific and technical information; the participation of the USSR in foreign exhibitions.

Cooperation with Socialist Countries. As is well known, back in 1955 the first agreement of the Soviet Union with socialist countries on a bilateral basis for cooperation in the field of the peaceful uses of atomic energy were concluded.

State Committee on the Use of Atomic Energy of the USSR. Translated from Atomnaya Énergiya, Vol. 31, No. 4, pp. 430-438, October, 1971.

The main goal in those years was the rendering of aid by the Soviet Union in the creation of the scientific and technical base necessary for them, in the training of personnel for the development of national atomic science and technology, in the construction of nuclear research reactors, accelerators of elementary particles, physical and radiochemical laboratories. As a result of such cooperation, national atomic centers were created in the People's Republic of Bulgaria, the German Democratic Republic, the Hungarian People's Republic, the Polish People's Republic, the Czechoslovakian SSR, the Socialist Republic of Rumania, and the Socialist Federated Republic of Yugoslavia, equipped with the most modern types of research reactors, accelerators of elementary particles, and other nuclear installations, aiding in conducting scientific research work at the most modern level. Soviet specialists, who transmitted their knowledge and experience to specialists in the brother countries, took part in the construction, assembly, adjustment, and starting of these installations.

The creation of atomic centers and their activity has promoted a broader development of scientific research, engineering developments, and subsequent organization of new institutes and nuclear installations in the socialist countries.

Enormous possibilities for peaceful cooperation have been vested in the socialist construction itself. These possibilities are determined by the complete harmony of interests of all the socialist countries, united by common goals. It is precisely therefore, that the forms of scientific cooperation have been expanded and improved with each passing year. This circumstance has found expression in the content of new agreements with the socialist countries enumerated above, with a further development of cooperation in the field of the use of atomic energy for peaceful purposes.

In January 1969, the opening of the atomic center of the republic of Cuba was held; it was created with the technical aid of the Soviet Union on the basis of the 1967 agreement.

All the agreements are based on premises of conducting joint scientific research work in the field of nuclear physics, solid state physics, the physics and technology of nuclear energy reactors, plasma physics, radiochemistry, the development and organization of the production of radioisotopic instruments, etc. An exchange of scientific and technical information, the accepting of apprentices, the holding of mutual consultations, the supply from the USSR and mutual supply to one another of special materials and equipment for conducting scientific research work are also being carried out.

The most important trend in the development of scientific and technical relations with the socialist countries is the cooperation in the field of atomic energetics. Agreements for the construction of atomic electric power plants provide for close mutual cooperation of scientists, engineers, and specialists participating in the development, planning, and installation of atomic electric power plants, the exchange of information and technical documentation on work and investigations performed, as well as the holding of bilateral consultations, associated with the creation of atomic electric power plants.

In accord with the agreement with the German Democratic Republic (1956), the Soviet Union has rendered technical aid in the construction of a 70 MW atomic electric power plant with a water–water type reactor in the region of Reinsberg. In 1956 the atomic electric power plant was put into operation, and it has already been operating successfully and reliably at the plant capacity for five years.

In July 1965, a new agreement was concluded for further expansion of cooperation in the installation and introduction into operation of atomic electric power plants with a total electric capacity of 2000 MW in the German Democratic Republic by 1980. The first stage in the implementation of this plan is the installation of the Nord-1 Atomic Electric Power Plant, with a capacity of 800 MW, consisting of two units with VVÉR type reactors. The first unit is planned for introduction into operation in 1973, the second in 1974. The startup of two units of the Nord-2 Atomic Electric Power Plant with analogous capacity is planned for 1977-1978.

Work on the preparation for startup of the first atomic electric power plant in Czechoslovakia, with a capacity of 150 MW, with a vessel-type reactor with gas cooling on natural uranium, is in the concluding stage. In the course of joint cooperation, Soviet and Czechoslovakian scientists, engineers, and specialists have successfully performed complex technical tasks, associated with the development and preparation of the entire complex of equipment for the atomic electric power plant. On the basis of the experience obtained, the industry of the Czechoslovakian SSR has developed the production of high-pressure vessels and other technological equipment.

In 1970 a new agreement for construction of two atomic electric power plants with a total electrical capacity of about 1700 MW, consisting of four units with reactor installations of the VVÉR type, by the Soviet Union in the Czechoslovakian SSR, beginning with 1977, was signed.

In 1966 agreements were also concluded on the installation of atomic electric power plants in the People's Republic of Bulgaria and the Hungarian People's Republic, with an electrical capacity of 800 MW, each of which consists of two units with VVÉR reactors, and in 1970 an agreement was made with the Socialist Republic of Rumania on the construction of an atomic electric power plant with a VVÉR-440 type reactor.

Thus, thanks to the wide and fruitful cooperation in the socialist countries, with the aid of the USSR, a large scientific and technical base in the field of atomic energetics has been or is being created, including scientific research and design and planning institutes, factories and enterprises, which can solve complex scientific and technical problems in this field.

The cooperation of the USSR with the socialist countries is not limited only to bilateral agreement.

In connection with the creation of the scientific and technical prerequisites, new problems have arisen associated with the utilization of the achievements of atomic science and technology in various branches of the national economy. The solution of such problems has required considerable facilities, extensive installations, instruments, apparatuses, and materials. The need has arisen for resolving certain very important scientific and technical problems through joint efforts.

In 1956, through the combined efforts of the socialist countries, the United Institute of Nuclear Research (OIYaI) was set up in Dubna to provide for joint technical and experimental investigations in the field of nuclear physics. And for 15 years this institute and its multinational group of scientific co-workers has already been conducting important and interesting investigations, the results of which are being utilized by all the countries participating in this institute. OIYaI is conducting many of its studies in collaboration with scientific research institutes of the member countries. The cooperation of OIYaI with the scientific institutes of the Soviet Union is bringing great profit to both countries.

An example of such close cooperation can be found in the joint scientific research work of OIYaI and the Institute of High-Energy Physics of the State Design Institute of Atomic Energy (GKIAE) of the USSR on the Serpukhov accelerator.

In 1970 an agreement was concluded between OIYaI and the State Committee for scientific and technical cooperation, according to which both sides were obligated to promote the further development of cooperation in the field of nuclear physics, ensuring maximum and effective utilization of the accelerators, research reactors, apparatuses for information processing, and other experimental and research installations available to them, as well as to create new equipment for these purposes. Bilateral agreements with a definition of the subject matter of research, periods for conducting experiments, etc., will be concluded between OIYaI and the institutes of the State Committee for the performance of these tasks.

Successful cooperation of Soviet scientists with scientists of the socialist countries is continuing within the framework of the permanent commission of the Council of Economic Mutual Aid for the use of atomic energy for peaceful purposes, formed in 1960, in the work of which delegations of the People's Republic of Bulgaria, the Hungarian People's Republic, the German Democratic Republic, the Polish People's Republic, the Socialist Republic of Rumania, the Czechoslovakian SSR, and the USSR are participating.

The goal of the commission is to aid the further development of multilateral economic and scientific and technical cooperation among the member countries of the Council of Economic Mutual Aid in the interest of the planned application of atomic energy.

The permanent commission is organizing the most important scientific research work and engineering developments in the field of reactor science and technology and atomic energetics, nuclear instrument construction, the production of isotopes and sources of nuclear radiations, the use of radioisotopic methods and apparatus, radiation safety and shielding techniques, the removal of radioactive wastes, and other interesting and important problems. For the better organization and coordination of work on the use of atomic energy and the satisfaction of the growing energy requirements of the member countries of the Council of Economic Mutual Aid, many of which do not have large resources of classical fuel, a special working group of specialists on problems of design, installation, and operation of atomic electric power plants has been created within the framework of the commission.

This group has developed a plan for the coordination of scientific and technical research, which has provided for the investigation of 37 topics, including 10 topics in reactor technology and nuclear energetics. The Soviet specialists have participated in the studies on:

the investigation and development of new energy reactors and the improvement of existing ones on thermal neutrons with an electrical capacity of more than 400 MW, especially reactors with water under pressure;

the investigation and planning development of an energy reactor on fast neutrons with various coolants, with an electrical capacity above 1000 MW;

the development and investigation of the technology of the production and regeneration of nuclear fuels;

investigation in the field of the production of new reactor materials, special equipment and means of radiation protection and safety of energy reactors, etc.

Soviet specialists are participating in modern scientific research work within the framework of the permanent commission of the Council of Economic Mutual Aid for the production and use of isotopes, labeled compounds, and radiation sources, in the field of nuclear physical instruments and radioisotopic apparatus, on problems of radiation safety and shielding techniques, the detoxification of liquid, solid, and gaseous radioactive wastes, the deactivation of contaminated surfaces, etc.

The member countries of the Council of Economic Mutual Aid are also cooperating widely in the field of nuclear instrument construction.

By now the total volume of production of articles of nuclear instrument construction in the member countries of the Council of Economic Mutual Aid is more than 65 million rubles, with an assortment covering more than 100 types of instruments and up to 1000 names. The total nomenclature of radioisotopic production covers more than 5000 preparations and radiation sources.

In 1970 a program of further strengthening of cooperation in the field of isotopic production for 1971-1975 was adopted at the 18th meeting of the commission. The program provides for complex solution of problems of specialization and cooperation, unification and standardization, scientific and technical collaboration, and improvement of the system of information.

Cooperation in the field of nuclear medicine, radiation processes, and installations is also being developed and improved.

The cooperation of the socialist countries in the field of the peaceful atom within the framework of the Council of Economic Mutual Aid is developing inseparably with the trend of establishment of concrete practical relationships the constant mutual search for means of solving new tasks and problems.

Cooperation with Developing Countries. Plans for the use of atomic energy for peaceful purposes are becoming a powerful stimulus for economic and technical progress of the developing countries. The implementation of this goal is largely promoted by the scientific and technical cooperation of these countries with the USSR on the basis of bilateral agreements and within the framework of the IAEA.

The Soviet Union, expanding cooperation with the developing countries, is striving to share its experience with the scientists of these countries to render technical assistance in the creation of natural scientific research centers and the training of local personnel for independent scientific research.

The Scientific Research Atomic Center created in 1961 close to Cairo, on the basis of an intergovernmental agreement of the USSR with the United Arab Republic, has substantially expanded the topics of joint studies included in the plan of investigation of thermophysics and plasma physics in recent years, and has also increased the volume of research on the use of radioactive isotopes.

The center includes an experimental water–water reactor with a capacity of 2000 kW, a highly stable electrostatic accelerator with an energy of 2.5 MeV, as well as shops and special laboratories for conducting research work in the field of nuclear physics, chemistry, metallurgy, and biology.

From 1964 up to the present time, about 100 scientific works have been conducted at the atomic center of the UAR. Soviet scientists have delivered more than 400 lectures to Egyptian specialists and have held many seminars. In 1963, the Middle East Regional Radioisotope Center was founded on the basis of the atomic center in Cairo, and 13 countries are participating in its work.

A center of nuclear research has also been created in Iraq with the aid of the Soviet Union, with a IRT-2000 type reactor, which was started up in 1968. The center has a division for the production of radioactive isotopes, divisions of physics, geology, radiation safety, biology, and agriculture, as well as repair and technical services. About 50 Soviet specialists were detailed to render aid in the construction, assembly, and adjustment of the equipment sent to Iraq.

In 1969 a protocol was signed for Iraq for further expansion of scientific and technical cooperation in the field of the peaceful use of atomic energy. Modernization of the research reactor, aimed at increasing the reactor power, is provided for.

In 1970 an agreement for cooperation, providing for mutual exchange of specialists and experts on the technology of nuclear energy reactors, the production and processing of nuclear fuel, the exploration and mining of nuclear raw material resources, investigations in various fields of atomic science and technology, the technology of dual purpose nuclear energy installations, etc., was concluded between the GKIAE of the USSR and the Atomic Energy Commission of Pakistan. The State Committee will each year send up to five stipends for the specialization of Pakistani scientists and their acquisition of practical experience in Soviet scientific research institutions and will render aid in the acquisition of the necessary equipment in the USSR.

Promoting the development of nuclear energetics of the Republic of India, the Soviet Union, in accord with the 1961 agreement, sent complete technical documentation to India for a fast neutron reactor with an electrical power of 50 MW. An exchange of delegations of specialists between the USSR and India is also being effected.

The Soviet Union is rendering considerable support to the developing countries, participating actively in the realization of the program of technical aid of the IAEA, the importance of which is increasing substantially in connection with the statement on the strength of the Nuclear Arms Limitation Treaty, since refusal to produce and acquire nuclear weapons permits developing countries to free themselves of large nonproductive expenditures and creates additional possibilities for the use of available resources for economic and social progress.

The Soviet Union has also placed at the disposal of the IAEA contributions of a total sum of 712,500 rubles in the form of shipments of installations, equipment, materials, instruments, and articles to developing countries.

The Soviet Union has provided the IAEA gratis with a special contribution in the form of equipment for four radiological centers: in 1967 in Morocco, in 1968 in Pakistan, in 1970 in Iraq; in addition, a resolution has been adopted for the construction of a center in Burma in 1972-1973.

The Soviet Government has adopted a resolution to increase the contributions of the USSR to the fund of technical aid of the IAEA in 1971 from 150 to 250 thousand rubles for the acquisition of equipment, instruments, and materials, as well as for the holding of conferences, scientific acquaintance visits, courses, and seminars in the USSR.

At the present time, all 20 stipends (up to one year each), provided in 1966 at the disposal of IAEA for the training of specialists from developing countries, members of the Agency, in the scientific research centers of our country, have practically been realized. In 1970, at the 14th session of the General Conference of the IAEA, the Soviet Union provided the Agency with another 25 stipends. The USSR has also confirmed its readiness to provide 10 stipends each year for the training of specialists for work on installations conducted with the aid of the USSR, or for conducting joint work within the framework of bilateral agreements with these countries.

In accord with the program of development of the United Nations Organization and the regular program of the IAEA, the Soviet Union has sent highly qualified consulting experts to the developing countries to render aid in the field of radiation safety, the irradiation of food products, the use of radioactive isotopes in hydrology, immunology, accelerator and reactor technology, nuclear physics, for the analysis of nuclear raw materials, etc.

Interest of the developing countries in participating in scientific acquaintance visits in the Soviet Union, attending week-long seminars, is increasing. In the last three years, seminars on the use of isotopes in industry, problems of handling radioactive wastes, radioisotopic methods of measurements (in vivo) in medicine, the use of isotopes and radiations in agriculture, and on standardization of radiation dosimetry have been organized at the request of the IAEA. Each group included about 25 foreign specialists.

Cooperation with Industrially Developed Countries. During the past period, an expansion of the cooperation of the USSR with industrially developed countries in the field of the peaceful use of atomic energy has been observed. The conclusion of agreements for cooperation with the United States, England, France, Canada, Italy, Sweden, Belgium, Holland, and Denmark has been an important step along the way of establishing closer working contacts in the field of atomic science and technology.

The same goals have largely been promoted by an exchange of delegations at the level of supervisors of national atomic commissions, as well as scientists and specialists for acquaintance and for conducting joint investigations on individual scientific problems.

In accord with the memorandum on cooperation in the field of the peaceful uses of atomic energy between the GKIAE of the USSR and the AEC of the United States, a mutual exchange of delegations on nuclear reactors, plasma physics, solid state physics, nuclear physics, problems of burial of radioactive wastes, etc. has been made. Since May 1970, a joint Soviet–American experiment on pion–electron scattering has been under way at the Institute of High-Energy Physics (IFVE, Serpukhov).

An exchange of delegations on industrial radiation problems and work on the use of energy beams for the heating of plasma has been agreed upon, and an exchange of scientists for conducting joint work in the field of plasma physics and controlled thermonulcear synthesis is being effected

In the development of the memorandum on cooperation, a protocol was signed in Washington in 1970 between the GKIAE of the USSR and AEC of the United States for conducting joint work in the field of high-energy physics on the accelerators of the IFVE (USSR) and the National Accelerator Laboratory (Batavia). Negotiations for conducting joint experiments on proton–proton small-angle scattering on the accelerator in Batavia are now under way.

In addition, on the basis of a mutual agreement, three stages of Soviet–American technical negotiations on the peaceful uses of nuclear explosions have been conducted in Vienna, Moscow, and Washington.

The agreement for cooperation of the atomic organizations of England and the Soviet Union, signed in 1961, now extends up to 1976. In accord with the agreement, an exchange of delegations has been held to foster an acquaintance with work on nuclear physics, plasma physics, accelerators, high energy physics, radiation safety, etc.

Agreement has also been achieved for cooperation in the field of energy reactors and reactor materials. A new stage has been the conducting of joint experiments at the I. V. Kurchatov Institute of Atomic Energy for the measurement of the electronic temperature of plasma on the Tokamak-3 installation with the aid of a laser beam.

Scientific and technical cooperation with France in the field of the peaceful uses of atomic energy is being implemented on the basis of the 1967 agreement, which was extended several times, expanding the base for exchange of delegations, information, and for conducting joint scientific measures.

A parallel agreement has been implemented between the GKIAE of the USSR and the Atomic Energy Commission of France, signed in 1966, for conducting joint scientific research work in the field of high-energy physics on the Soviet proton accelerator with energy 70 GeV in Serpukhov, using the French Mirabel liquid hydrogen bubble chamber with a useful volume of 6000 liters. We can now note with satisfaction that beginning with the exchange of delegations for acquaintance with scientific centers, our cooperation has progressed to the conducting of large joint scientific experiments. The signing and realization of this agreement represents not only an outstanding contribution to the development of high-energy physics, but also an example of the fruitful cooperation of two countries with different social structures in the spirit of friendship and mutual understanding.

In connection with the announcement of the Soviet Union at the 12th Session of the General Conference of the IAEA of readiness to enrich the uranium raw materials of interested countries up to 2.5-5% with respect to U^{235}, in May, 1971, a contract was signed in Paris to provide the French side with the service of enrichment of French uranium ore at Soviet enterprises under mutually profitable conditions.

According to the agreement between the GKIAE of the USSR and the European Organization of Nuclear Research (CERN) in 1967, such equipment for the Serpukhov accelerator as a system of rapid discharge of a beam of protons, a high-frequency separator of elementary particles, as well as various electronic apparatuses is being developed with the active participation of Soviet specialists in CERN. After the delivery

of this equipment to Serpukhov, a large group of CERN scientists will come to conduct joint scientific work on the accelerator. Thus, after the assembly and adjustment of this equipment in the IFVE, a unique complex, including a multistill liquid hydrogen Mirabel chamber and the largest charged-particle channel in the world will be created through the joint efforts of scientists of different countries. Soviet institutes, scientific research organizations of France and the member countries of CERN will participate in the processing of information from the Mirabel chamber, obtained in the form of stereophotographs.

Cooperation with the Canadian state organization "Atomic Energy of Canada, Ltd." on the basis of the 1964 agreement with the National Committee on Nuclear Energy of Italy, the reactor center of the Netherlands, the Comissariat of Atomic Energy of Belgium (since 1965), and the Atomic Energy Commission of Denmark (1968) is also under way.

In recent years the cooperation of the USSR with Sweden in the field of atomic science and technology has been appreciably expanded.

In accord with the agreement fixed in the Soviet–Swedish communiqué of 1968, a 30 year agreement between the governments of the USSR and Sweden for cooperation in the field of the peaceful uses of atomic energy was signed in Moscow, 1970. The agreement provides for expansion of the volume of cooperation in comparison with the protocol for cooperation in the field of the peaceful uses of atomic energy signed in 1968, between the GKIAE of the USSR and the Swedish Royal Academy of Engineering Sciences, according to which there is only an exchange of delegations of scientists and scientific and technical information.

The indicated agreement provides for mutual supply of equipment, including nuclear reactors and fuel for them, materials and special nuclear materials on a commercial basis, services for enrichment of uranium at the enterprises of the USSR, and any materials provided according to the agreement will be used only for peaceful purposes.

Contacts with the German Federated Republic are being established: in 1970 negotiations on prospects of cooperation between groups of experts of the GKIAE of the USSR and the Federal Ministry of Education and Science of the German Federated Republic were held in Moscow; in April, 1971 the USSR sent a delegation of specialists to the German Federated Republic in the field of high-energy physics; in June, 1971 Soviet specialists on problems of thermonuclear synthesis were sent to the German Federated Republic.

In 1969, a protocol for the construction of an atomic electric power plant with a 440 MW capacity possessing a reactor with water under pressure, in Finland with the technical aid of the USSR, was signed between the governments of the Soviet Union and Finland. Readiness of the station for industrial operation will be guaranteed before June 30, 1976.

Participation of the USSR in the Activity of International Organizations. As has already been noted, the USSR is rendering great aid to developing countries within the framework of the IAEA – an organization specially founded for promoting the use of atomic energy for peaceful purposes and progress, and the implementation of cooperation in this field.

Since the creation of the agency, the Soviet Union has participated actively in its activity, aiding in every way to fulfill the tasks placed before it.

During the period that has elapsed, Soviet specialists have substantially expanded participation in conferences, symposia, meetings of experts, working groups, and seminars held by the agency and has presented a large number of reports and communications on the latest achievements of Soviet atomic science and technology. In 1970 alone, about 100 Soviet reports were presented. Such participation is permitting the establishment and reinforcement of relations with foreign scientists and is sharing the rich experience accumulated by Soviet atomic science and technology.

Among the measures that have been taken in the USSR, we should mention the meeting of the International Committee for Nuclear Data (Moscow, 1967); the Third Conference on Investigations in the Field of Plasma Physics and Controlled Thermonuclear Reactions, at which more than 250 foreign scientists from 20 member countries of the IAEA were present (Novosibirsk, 1968); the meeting of experts on investigations of the structure of the nucleus, problems of the conservation and transplantation of bone marrow, the fixation of radioactive wastes in bitumen; the meeting of the international group of communications on thermionic emission transformation, etc.

Frequent meetings in the State Committee with co-workers of the Secretariat of the IAEA are also practiced for the discussion of various problems associated with the participation of the USSR in the activity of the agency.

The Soviet specialists have participated actively in the regulated activity of the IAEA for the development of standards, instructions, rules and recommendations, chiefly on problems of nuclear safety.

At the present time, Soviet research institutions are carrying out 11 contracts and agreements (six of them on a nonpaying basis) with the IAEA on technical aspects of guarantees, removal of radioactive wastes, radiospectroscopic investigations, on the use of induced mutations in the selective breeding of plants, etc.

The Soviet Union also attributes great significance to such important aspects of the activity of IAEA as the implementation of control functions in connection with the Nuclear Arms Limitation Treaty, the use of nuclear explosions for peaceful purposes, the international system of nuclear information, and the coordination of work in the field of thermonuclear synthesis.

The Soviet delegation has participated actively in the work of the Committee on Guarantees, by which concrete recommendations have been developed on the structure and content of agreements on control, required by the treaty. Practical possibilities have thereby been opened up for negotiations and the conclusion of agreements on control between nonnuclear states participating in the treaty and the IAEA.

Participating in measures of the IAEA on the investigation of the problem of peaceful nuclear explosions, Soviet specialists have shared their experience in conducting work and development of technical projects for the use of nuclear explosions in various regions of the USSR national economy. At the 14th Session of the General Conference of the IAEA, the Soviet delegation shared with the agency motion picture films on the use of atomic explosions for peaceful purposes (the elimination of a burning gas fountain and the creation of an artificial water reservoir) and a collection of scientific materials on certain studies conducted by the Soviet Union in this field.

Considering the broad international exchange of information as a means of accelerating scientific, technical, and economic progress, the Soviet Union has been one of the initiators of the creation of the first international system of nuclear information in the world, within the framework of the IAEA; it began operation in April of 1970. Leading Soviet specialists participated in the activity of the working agencies of the IAEA, occupied with the development of the organizational and technical principles of this system.

Since the creation of the International System of Nuclear Information, the Soviet Union, making use of its self-imposed obligations, each month has made a regular deposit of Soviet materials falling under the scope of its subject matter into the system.

Aiding the further improvement of this system every way, we believe that its successful functioning will aid in the development of atomic science and technology in the developing countries, and also permit industrially developed countries to exchange scientific and technical information more effectively on the basis of multilateral cooperation.

The Soviet Union supported the intention of the IAEA to found an international council on thermonuclear research within the framework of the agency, in order to improve international coordination of work in the field of the creation of the thermonuclear reactor The USSR is represented in this council by Academician L. A. Artsimovich.

The International Center of Theoretical Physics in Trieste, founded in 1964 by the agency, the activity of which is now controlled jointly with UNESCO, is promoting the further development of all fields of theoretical physics through the training of young physicists, especially from the developing countries, and conducting scientific research and forums.

Eminent Soviet scientists have participated in the measures of the center, delivering lectures, conducting scientific research work (meeting of the working group on plasma physics, 1970), and participating in the work of symposia (International Symposium on Problems of Modern Physics, 1968).

The international coordination, implemented within the framework of IAEA is greatly promoting the expansion of economic and scientific and technical relations among states, as a result of which duplication in scientific research work is being eliminated, with a substantial saving of time and facilities.

The Soviet Union is participating actively in the work of the scientific committee of the United Nations Organization on the Effects of Atomic Radiation, which has prepared a number of reports for the UN General Assembly on radioactive fallout as a result of nuclear weapons tests, on the action of natural and artificial radiation on the environment, on radioactive pollution of the environment, etc. The evaluations and conclusion drawn by the committee have played a positive role in the struggle for the cessation of nuclear tests in the atmosphere, cosmic space, and underwater, and has promoted an expansion of knowledge of the levels and action of atomic radiation arising from all sources.

In striving to aid the expansion of international cooperation in the field of standardization, Soviet specialists are participating actively in the activity of the technical Committee 45 ("Electrical Measuring Apparatus for the Registration of Ionization Radiations"), of the International Electrotechnical Commission, its subcommittees and working groups, as well as technical Committee 85 ("Atomic Energy") of the International Organization of Standardization.

The Soviet Union has played host (Moscow, 1969) to a regular meeting of Technical Committee 45, in which 60 foreign specialists participated, at which questions associated with the classification of terminology, the unification of radioisotopic apparatus, methods of testing, etc. were discussed.

Participation in International Fairs and Exhibitions. An important role in the strengthening of international cooperation and the development of economic and trade relations of the USSR with foreign countries is played by Soviet exhibits, held both abroad and within the country, and reflecting the achievements of the Soviet Union in the field of the peaceful uses of atomic energy.

The exhibits "Atoms for Peace," which have been demonstrated in many countries of Asia, Africa, Europe, North and South America, have enjoyed invariable success.

At the International Exhibition in Montreal EXPO-67, in the exposition of the "Atoms for Peace" section, the wave from the first Soviet atomic electric power plant in the world to the modern industrial-type atomic electric power-plants, as well as the success of the USSR in many other regions of utilization of atomic energy, were demonstrated.

In 1967 anniversary exhibitions "Scientific and Technical Achievements of the USSR," at which the scientific achievements of our country in 50 years and their application in the national economy, were demonstrated in the Hungarian People's Republic and the Socialist Republic of Rumania.

The Soviet Union participated in the International Atomic Industry Fair NUCLEX-60 in Basel (Switzerland). Models of atomic reactors, isotopic current sources, accelerators for industrial purposes, plasma installations for the cutting of metals, laser setups and other objects were represented in the Soviet pavillion. During the exhibition, a scientific and technical conference of representatives from 22 countries of the world, in which Soviet specialists also participated, was held.

Soviet atomic science and technology was also widely represented at the International Exhibition EXPO-70 in Osaka. The basic content of the Soviet pavillion was a demonstration of the achievements of science and technology, with which the Soviet people have come to the glorious anniversary of the organizer of the soviet state, V. I. Lenin.

Our pavillion at the exhibitions here in Geneva, coordinated with the holding of the Fourth Geneva Conference on the Peaceful Uses of Atomic Energy, also tells about the successes of Soviet atomic science and technology.

In only 16 years the Soviet Union has organized 152 exhibits on atomic topics, including foreign exhibitions in 45 countries of the world. The number of visitors to these exhibits abroad – more than 80 million persons – is evidence of the great interest that is being shown in the success of Soviet atomic science and technology.

Since the Third International Conference on the Peaceful Uses of Atomic Energy, the basic trends of work in this field have received further development, and the role of atomic energy in the life of mankind has increased.

The Soviet Union, as a mighty nuclear power, is interacting with other states, not from the standpoint of war, but from the standpoint of peace, on which our policy of peaceful coexistence, developed by V. I. Lenin, is based.

The foreign policy of our Party and Government is directed toward the further manifold improvement and expansion of economic and scientific and technical relations with all the countries of the world under conditions of mutual profit and equality in the interest of peace. The international cooperation of the USSR in the field of the peaceful uses of atomic energy serves these same goals.

ULTRASTRUCTURE OF BACTERIAL VIRUSES

By **A. S. Tikhonenko**
Institute of Molecular Biology
Academy of Sciences of the USSR, Moscow

During the past twenty-five years, bacteriophagy has developed rapidly as a cornerstone of modern biology and has attracted eminent specialists from various scientific disciplines. Progress made in many of the fundamental problems in modern biology is largely contingent upon the study of phages. In recent years phages have provided the primary model for the study of basic problems in molecular biology. Phages served as the model for Nobel-Prize-winning classical investigations on the fine structure of the gene, deciphering the genetic code, and studying the mechanism of transmission of information for protein synthesis.

Containing up-to-date information on the ultrastructure of bacterial viruses and their development in the bacterial cell, this volume provides a foundation for future progress in the biological sciences. It examines electron-microscopic techniques, generously illustrated with original electron micrographs. Presenting a morphological classification of phages, the book also covers changes in the structure of bacteriophages under the influence of chemical and physical agents. the functional role of the individual structural elements of the phage particle during infection, and the isolation of individual structural proteins of the phage in a pure form. The volume contains the results of studies on the properties of contractile phage proteins and the localization of enzyme (phosphatase) activity in the structural components of the phage particle. The book gives a comprehensive account of phages of thermophilic actinomycetes.

OF INTEREST TO: molecular biologists, virologists, microbiologists, biophysicists, and biochemists working with bacterial viruses, as well as students of medicine and advanced biology.

Approx. 260 pages PP May 1970 **$19.50**

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AGING IN CELL AND TISSUE CULTURE

Proceedings of the European Tissue Culture Society's Annual Meeting, held at the Castle of Zinkovy Czechoslovakia, May 7-10, 1969

Edited by **Emma Holečková**
Institute of Physiology, Czecnoslovak Academy of Sciences Prague, Czechoslovakia

and **Vincent J. Cristofalo**
Associate Member Wistar Institute of Anatomy and Biology Philadelphia, Pennsylvania

Biological and medical advances have extended man's potential life-span to the point at which the phenomena of aging assume primary importance as a subject for research. Studies of the smallest unit of living organisms—the individual cell—and of the tissues composed of such cells have already raised heated controversies:

- How relevant are studies of proliferating cell cultures to aging *in vivo*?
- Is the senescence observed in cell cultures an intrinsic, programmed function of the cell, or is it caused by nutritional deficiencies or the trauma of cultivation?
- What causes the decline in the proliferative capacity of cultured cells—the number of cell generations or the metabolic lifetime of the cells examined?
- Is the limited life-span of normal vertebrate cells *in vitro* species-specific?

This review of present knowledge on the morphological and physiological changes in cultured cells during aging considers the questions of aging in culture and of cells cultured from donors of different ages. It presents, furthermore, the first review of the experimental evidence from European and American laboratories concerning the Hayflick-Moorhead hypothesis that aging is an obligatory feature of diploid cell growth.

Included among the contributions to this volume are studies on

- Heterogeneity of cell size and chromosome number in relation to explantation
- Ability of cells from donors of different ages to migrate and multiply
- Metabolic changes accompanying aging *in vitro*, such as nucleic acid metabolism, respiration, glycolysis, lipid metabolism, and the activities of various enzymes
- Dynamic morphology of cell death

OF SPECIAL INTEREST TO: research workers in the life sciences; particularly those in biochemistry, molecular biology, biophysics, zoology, and medicine.

163 pages 1970 **$12.50**

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